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# Geological Disposal

## Generic Operational Safety Case main report

December 2010

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## Preface

We, the Nuclear Decommissioning Authority (NDA) have been charged with implementing the UK Government's policy for the long-term management of higher activity radioactive waste by planning, building and operating a geological disposal facility (GDF). Our Radioactive Waste Management Directorate (RWMD) is working on a programme to implement geological disposal in the UK. The UK has accumulated a legacy of radioactive waste from electricity generation, reprocessing, fuel manufacturing, defence activities and other industrial, medical, agricultural and research activities. Radioactive wastes continue to be produced from these activities. Some of these wastes will remain hazardous for thousands of years. The development of new nuclear power plants in the UK would lead to the generation of wastes similar to those already in existence.

The UK Government has undertaken a wide-ranging consultation on the best means of dealing with higher activity radioactive wastes. This led, in October 2006, to the UK Government deciding that geological disposal, preceded by safe and secure interim storage, is the way forward for the long-term management of these wastes[1]. The UK Government published a White Paper in June 2008 that explains the way forward for implementing a national GDF. The first step is for the UK Government to seek expressions of interest from communities that may wish to consider participating in a site selection process. This is to be followed by desk-based studies of the available areas.

An important part of our preparatory work is for us to set out an approach for assessing the safety of the disposal system. We are setting out the approach we propose in an integrated set of documents under the collective title of the 'generic Disposal System Safety Case' (DSSC). The generic DSSC presents methods, evidence and arguments concerning the safety of the transport of waste to a GDF, and the construction, operation and long-term safety of a GDF for UK higher activity radioactive wastes.

This document is the generic Operational Safety Case main report (OSC main report). It considers the safety of a GDF during the construction and operation of the facility through to sealing and closure. The generic OSC main report has been written to demonstrate the safety of operations at a GDF and to show compliance with relevant regulatory design targets and limits. Two companion documents address; (i) the safety of transport of waste to a GDF (the generic Transport Safety Case main report) and (ii) environmental safety of a GDF during the operational period and after closure (the generic Environmental Safety Case main report).

As part of the generic DSSC suite of documentation, we are also publishing an overview report for a wider readership, and a series of safety assessment reports and status reports at a more detailed level on key topics relevant to the safety of a GDF, as well as other supporting reports. The OSC – and the overall DSSC – will be developed in phases, alongside the development of a GDF itself, over a period lasting several decades. This report provides a "generic" (i.e. not specific to any UK site) consideration of the Safety Case because it supports the first step of the site selection process and no specific sites for a GDF have yet been identified.

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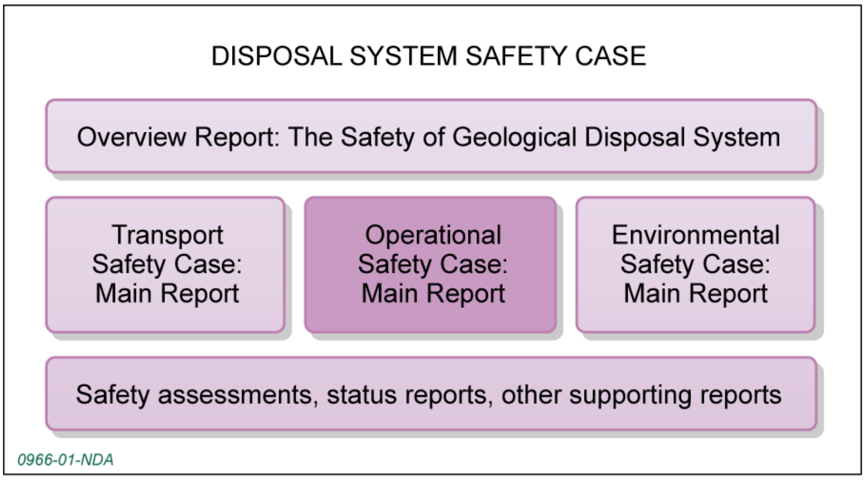
[1] Throughout this document, the term UK Government includes all relevant departments and devolved administrations, but excludes the Scottish Government, which does not support geological disposal – the Scottish Government advocates interim near-surface, near-site storage for higher activity radioactive wastes.




The generic OSC main report explains in principle why we have confidence in the construction and operational safety of a GDF, and the approach to developing the necessary safety case to demonstrate that confidence. Some illustrative safety assessment calculations are included in the report, based on work documented in the underpinning safety assessments and status reports.

Iterative development of the DSSC provides a management tool for use in the progressive development of research, site characterisation and engineering design programmes that respond to the evolving information needs and outputs of the safety case. We also expect the DSSC to be a continuing focus for dialogue and consultation with regulators and stakeholders with an interest in safety. We have produced this report now, mainly to provide information on our methodology for iterative development of the safety case, in order to obtain feedback on the approach, prior to undertaking more site-specific analyses. While producing the generic DSSC at this early stage, we have taken care not to make any safety claims which might prove to be unachievable as the site-specific safety case is developed.

While specialist regulators (principally the Health and Safety Executive) are the primary audience for this report, we have aimed to use plain English as far as possible, to encourage wider dialogue on the generic OSC and on the generic DSSC overall.

We will update the OSC at appropriate stages for a GDF development programme: site selection, site characterisation, planning approval, construction, waste emplacement, sealing and closure, and withdrawal of control over the GDF site. This approach is consistent with a staged development and approval process, as advocated by our regulators.

## Executive summary


The Nuclear Decommissioning Authority (NDA), through its Radioactive Waste Management Directorate (RWMD), is working on a programme to implement a geological disposal facility (GDF) for higher activity radioactive wastes, as set out in the UK[2] Government's Managing Radioactive Waste Safely (MRWS) White Paper.

The White Paper defines a Baseline Inventory of materials that may need to be managed through geological disposal: high level waste (HLW), intermediate level waste (ILW), some low level waste (LLW) unsuitable for near-surface disposal, spent fuel (SF), depleted natural and low-enriched uranium (DNLEU), highly-enriched uranium (HEU) and separated plutonium (Pu).

We are currently in the first phase of the programme, and have produced a generic Disposal System Safety Case (DSSC). The generic DSSC explains and assesses the safety and environmental implications of all aspects of geological disposal of higher activity radioactive waste in the UK.

The generic DSSC addresses the following:

- transporting the waste to a GDF – the safety arguments and assessment of this are presented in the **generic Transport Safety Case (TSC)**
- construction and emplacement of waste within a GDF, subsequent storage and eventual backfilling, decommissioning and closure – presented in the **generic Operational Safety Case (OSC)**
- the environmental safety of a GDF during the operational period and after closure of the facility – presented in the **generic Environmental Safety Case (ESC)**.

The generic OSC presents the safety case for construction, operations, decommissioning and closure of the GDF and is supported by a number of more detailed assessment reports. These reports are in turn supported by a number of specifically commissioned reports, as well as a great deal of wider scientific literature.

### *Context and objectives*

At the current stage in the MRWS process, prior to the identification of potential sites, we cannot produce an OSC using site-specific information. The location and its geology will determine key characteristics of the GDF such as the dimensions of the underground disposal vaults where the waste packages will be emplaced. Such characteristics will have an impact on waste package handling and emplacement operations.

In developing the generic DSSC, we have reported our work on operational safety in the form of a generic OSC in order to:

- present safety information in a way that informs the regulators and the wider stakeholder community in a clear and concise manner to facilitate their engagement in the developing safety arguments
- provide a foundation for desk-based studies under the MRWS site assessment process
- provide the basis for the safety case documentation that, in due course, will be required to meet regulatory Site Licence Conditions
- support the Letter of Compliance disposability assessment process (which indicates to a waste packager that a proposed waste package is compliant with the relevant packaging criteria and disposal safety assessment).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[2] The implementation of geological disposal does not apply in Scotland

The generic OSC main report has been structured as a top-tier operational safety report. It is intended:

- to present the results of our operational safety assessment work
- to provide assurance that, in due course, we will be able to make a safety case for a GDF that will satisfy all regulatory requirements
- to explain how we will protect operators and members of the public from hazards arising during normal and fault conditions
- to present the safety assessment methodology we have employed to identify and analyse potential hazards during normal operations and under fault conditions
- to provide clear references to the documents containing more detailed information
- to identify our forward programme of work for developing and improving our OSC.

The generic OSC will provide a platform that will be updated once potential candidate sites have been identified through the MRWS site selection process. As the programme evolves, the OSC will also be used to update the basis for packaging advice to waste producers and to inform choices between GDF design options.

### *Safety strategy*

We have developed a Disposal System Specification (DSS) which sets out what is required of the GDF and provides the basis for the engineering designs that will underpin the generic DSSC.

In addition, we have developed a set of principles for safety, environmental protection, security and safeguards which apply from the earliest stages of the design process. They have been developed to conform to relevant good practice and to reflect the expectations of the regulators. Their purpose is to provide a basis for the design and assessment of geological disposal facilities and associated transport systems. The principles require designers to build safety features into the design which complement each other and contribute to overall safety by, for example, providing several lines of defence against the realisation of potential hazards.

The approach that we have taken, until such time as site-specific information becomes available, is to select three generic geological environments, encompassing typical UK geologies:

- higher strength rocks
- lower strength sedimentary rock
- evaporites.

Six illustrative geological disposal concept examples have been chosen based on these generic environments, in order to support the generic DSSC.

Conceptual designs have been developed for these examples based on extensive R&D carried out in the UK and by our sister organisations overseas. However, this does not mean that any of these designs will necessarily be used in the selected site.

It is not appropriate to carry out detailed design optioneering on designs that are purely illustrative, although pertinent findings from our example safety assessments will be fed through to the next design stage where appropriate.

Safety assessment involves the identification of all the potential hazards associated with a nuclear plant and the demonstration that they are either precluded by suitable design features or, where this is not possible, adequately controlled. There are two key aspects to the assessment process. They are:

- the determination of a full set of potential faults and the evaluation of consequences and risks posed by these faults in order to ensure that they meet the relevant safety limits and targets;
- the demonstration that the engineered design is to a high standard and robust against potential faults such that the associated risks are restricted to levels that are As Low As Reasonably Practicable (ALARP).

At this early stage it is not possible to produce a comprehensive assessment. Rather we are developing our processes and methodologies for the production of safety cases and have carried out example assessments at this stage as an illustration of our approach. Our Radiological Protection Policy Manual describes the acceptability criteria we will use to judge the results of our safety analysis.

As our design and safety case work progresses we will develop appropriate safety documentation in line with major programme stages: site selection, site characterisation, planning approval, construction, operation, decommissioning, sealing, closure and eventual withdrawal of control. We are producing a Safety Case Manual, which will describe the detailed procedures and methodologies we will use to prepare these safety documents.

We are developing a Permissions Schedule identifying the various safety case documents and associated regulatory submissions against the MRWS site selection timeline.

The DSSC will in future provide the basis against which packages will be assessed through the Letter of Compliance disposability assessment process. This will provide an important confidence building component to the DSSC as the generic assessments are tested against the characteristics of “real” packages before they are manufactured by industry.

### *Assessment basis*

A Baseline Inventory of radioactive waste and materials for geological disposal is defined in the UK Government’s MRWS White Paper: high level waste (HLW), intermediate level waste (ILW), some low level waste (LLW) unsuitable for near-surface disposal, spent fuel (SF), depleted natural and low-enriched uranium (DNLEU), highly-enriched uranium (HEU) and separated plutonium (Pu). SF, DNLEU, HEU and Pu are not currently considered wastes but may be so in the future.

However, the Baseline Inventory does not contain sufficient detail to provide a basis for the designs and assessments underpinning the generic DSSC. We have therefore developed a Derived Inventory Reference Case that converts the Baseline Inventory data into the form required for design and safety assessments.

We have also defined an Upper Inventory based on alternative scenarios for waste arisings, including consideration of additional wastes and spent fuel from the operation and decommissioning of new build power stations.

The six illustrative disposal concept examples and the Baseline Inventory form the basis for the safety assessment presented here. The concepts include a set of waste package types that are suitable for the full range of wastes and materials identified.

The physical containment and form of radioactive waste also play a significant role in the safety considerations for operations at a GDF and so appropriate standards and specifications for the packaging of wastes have been developed. This provides assurance that waste packages currently being manufactured, will be compatible with the safe operation of the GDF. Packaging requirements are detailed in the suite of packaging specifications and supporting documents.

The safety assessment of the illustrative designs will be used to:

- support the scoping of the impacts of a GDF
- support the development of the disposal system specification, engineering designs and safety case methodology
- support prioritisation of the R&D programme.

### *Safety assessment of a GDF*

Operational safety is concerned with the safe handling and care of waste packages, from receipt at a GDF site and transfer underground, through to emplacement, backfilling and closure operations, together with the decommissioning of facilities no longer required. Our current assessment work focuses primarily on package handling and emplacement operations. Backfilling, decommissioning and closure operations will be considered more thoroughly in future assessments when more design detail is available.

The generic OSC considers potential radiological impacts to workers and the public, from both normal day-to-day operations and from fault conditions during the operational period. This is done in order to ensure that the GDF is designed in such a way as to reduce routine exposure to within regulatory limits and, so far as reasonably practicable, below those limits. Similarly, the design is scrutinised from the earliest stages to identify potential hazards and the faults which might cause them to be realised. Our safety specialists and designers work closely together during the development of the design in order to eliminate as many potential faults as possible by the provision of suitable design features. Where this is not possible, sufficient protection and safety systems will be incorporated in the design, to ensure that the consequences and risks associated with any fault condition are restricted so far as reasonably practicable and are within acceptable limits.

Suitable methodologies have been developed for carrying out safety assessment work which will be further refined as the design and safety work progresses.

Since there is not, at present, a site-specific design on which to carry out a detailed fault analysis, we have chosen, by way of example, to trial the methodology using two of the illustrative concept examples. For this work, we have chosen to analyse the illustrative concept examples for higher strength rock since this allows the use of slightly more detailed design features and assumptions developed for previous generic ILW/LLW and HLW/SF concepts. The generic OSC provides the results of this analysis and discusses the differences in safety performance we anticipate in the designs for the other two geologies.

The generic OSC provides a preliminary view on the safety implications of maintaining the GDF open for an extended period of monitoring and retrievability. The generic OSC also comments on the expected impact of the Upper Inventory.

### *Conclusions and forward programme*

The findings in this report provide confidence that it will be possible to develop a GDF design and mode of operation that will be capable of meeting modern safety standards for nuclear facilities in all three illustrative geologies.

In addition, we have considered safety during construction of a GDF, with preliminary hazard identification studies being undertaken. No hazards have been identified that cannot be adequately protected against. This gives us confidence that construction risks can be demonstrated to be acceptably low.

Once we start to develop site-specific designs, we will undertake more detailed hazard identification studies which will enable us to refine our fault schedule and future safety assessments.

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## List of abbreviations and acronyms

| | |
|---|---|
| AGR | Advanced Gas-cooled Reactor |
| ALARP | as low as reasonably practicable |
| BSL | basic safety level |
| BSO | basic safety objective |
| CoRWM | Committee on Radioactive Waste Management |
| DBA | design basis accident |
| DBF | design basis fault |
| DCTC | disposal canister transport container |
| Defra | Department of Environment, Food and Rural Affairs |
| DNLEU | depleted, natural and low enriched uranium |
| DSFS | Disposal System Functional Specification |
| DSS | Disposal System Specification |
| DSSC | Disposal System Safety Case |
| DSTS | Disposal System Technical Specification |
| EA | Environment Agency |
| ESC | Environmental Safety Case |
| GDF | geological disposal facility |
| GOSA | Generic Operational Safety Assessment |
| GWPS | Generic Waste Package Specification |
| HAZOP | hazard and operability study |
| HEPA | high efficiency particulate in air |
| HEU | highly enriched uranium |
| HLW | high level waste |
| HSE | Health and Safety Executive |
| IAEA | International Atomic Energy Agency |
| ILW | intermediate level waste |
| IP2 | Industrial Package Type 2 |
| IRRs | Ionising Radiations Regulations 1999 |
| LLW | low level waste |
| LoC | Letter of Compliance |
| MBGWS | Miscellaneous Beta Gamma Waste Store |
| MHSW | Managing Health and Safety at Work Regulations |
| MRWS | Managing Radioactive Waste Safely |
| ND | Nuclear Directorate |
| NDA | Nuclear Decommissioning Authority |
| NII | Nuclear Installations Inspectorate |

| | |
|---|---|
| NRVB | Nirex reference vault backfill |
| OSC | Operational Safety Case |
| PGRC | Phased Geological Repository Concept |
| PSA | probabilistic safety assessment |
| Pu | Plutonium |
| PWR | Pressurised Water Reactor |
| REPPIR | Radiation (Emergency Preparedness and Public Information) Regulations 2001 |
| RF | Release Fraction |
| RPPM | Radiological Protection Policy Manual |
| ROSA | Repository Operational Safety Assessment (Toolkit) |
| RSA | Radioactive Substances Act |
| RWMD | Radioactive Waste Management Directorate |
| SAA | Severe Accident Analysis |
| SAPs06 | Safety Assessment Principles (2006 Edition) |
| SEPA | Scottish Environment Protection Agency |
| SF | Spent Fuel |
| SFR | Safety Functional Requirement |
| SILW | Shielded Intermediate Level Waste |
| SWTC | Standard Waste Transport Container |
| TBM | Tunnel Boring Machine |
| TSC | Transport Safety Case |
| U | Uranium |
| UILW | Unshielded Intermediate Level Waste |
| UK RWI | UK Radioactive Waste Inventory |
| WAGR | Windscale Advanced Gas-Cooled Reactor |
| WIPP | Waste Isolation Pilot Plant |
| WPTF | Waste Package Transfer Facility |
| WVP | Waste Vitrification Plant |

# 1 Introduction

We, the Nuclear Decommissioning Authority (NDA), have responsibility for implementing a Geological Disposal Facility (GDF) in the UK, supporting the UK Government and devolved administrations in its commitments, as set out in the Managing Radioactive Waste Safely (MRWS) White Paper [1]. The MRWS White Paper includes the stages in a GDF site selection process that would lead to the identification of sites for desk-based studies, followed by surface investigations at candidate sites, and leading ultimately to the identification of a preferred site. The White Paper also defines the materials that may need to be managed through geological disposal. It defines a Baseline Inventory of high level waste (HLW), intermediate level waste (ILW), some low level waste (LLW) unsuitable for near-surface disposal, spent fuel (SF), depleted natural and low-enriched uranium (DNLEU), highly-enriched uranium (HEU) and separated plutonium (Pu). SF, DNLEU, HEU and Pu are not currently considered wastes but may be so in the future. We have also defined an Upper Inventory that includes wastes that may arise from new nuclear power stations.

Our Radioactive Waste Management Directorate (RWMD) is working on a programme to implement geological disposal in the UK. In the future, it is envisaged that RWMD will be established as a subsidiary of the NDA responsible for delivery of a GDF. RWMD is currently operating under voluntary scrutiny by our regulators as a “prospective Site Licence Company” (SLC) in order to demonstrate and develop the competences required of a future holder of a regulatory authorisation and nuclear site licence.

We are currently in the first phase of our programme, undertaking preparatory studies, and during this phase we have produced a generic Disposal System Safety Case (DSSC). The generic DSSC explains and assesses the safety and environmental implications of all aspects associated with the geological disposal of higher activity radioactive waste in the UK. The generic DSSC covers three main areas; for each we have prepared a separate safety report:

- transporting the waste to the disposal facility – the safety arguments and assessment of this are presented in the **generic Transport Safety Case main report (TSC)** [**2**]
- construction and emplacement of waste within a GDF, subsequent storage and eventual backfilling, decommissioning and closure – presented in the **generic Operational Safety Case main report (OSC)**
- the environmental safety of a GDF during the operational period and after its closure – presented in the **generic Environmental Safety Case main report (ESC)** [**3**]

When integrated, the generic TSC, OSC and ESC, and their supporting documents, comprise the generic DSSC, an overall statement of the safety of the complete disposal system. The hierarchy of the generic DSSC is shown in Figure 1.

This document, the generic Operational Safety Case (OSC) main report, provides an overview of the claims, arguments and evidence for safety, contained in a number of supporting assessments, which together comprise the generic OSC. These documents are supplemented in turn by a number of specifically commissioned reports, as well as a great deal of wider scientific literature. The generic OSC documentation is discussed in more detail in Section 1.2.1.

## 1.1 Scope of the generic OSC

This generic OSC is not site-specific and so we have used illustrative concept examples to examine the safety aspects of a GDF and the radioactive waste packages with respect to both normal operations and potential faults. The generic OSC focuses on the activities in the operational phase during which radioactive waste packages will be received on site, transported underground and emplaced in the disposal areas. Issues relating to possible retrieval of emplaced packages are also discussed.

The generic OSC also considers safety issues associated with the construction and commissioning of the above and below ground facilities, including the separation of construction and waste emplacement activities once a GDF becomes operational.

Operations such as backfilling, closure and decommissioning of the GDF have not been considered in any depth at this stage because there is insufficient design detail to support meaningful analysis. However, since these operations do not involve the physical movement of packages, we expect any contribution to risk from operations will be small. Our strategy for addressing decommissioning and closure operations within the design is discussed later in this report. As we develop our designs further we will assess the risks posed by these operations in detail.

Similarly, low risk surface facilities, such as those used for monitoring and handling of incoming waste packages, and auxiliary buildings such as laboratories and maintenance facilities, will be assessed when we have more design detail. We will also address the rail and road systems used to transport the waste packages across the site.

The generic OSC takes over from the generic Transport Safety Case (TSC) when the package is cleared for loading onto an internal transport vehicle for transfer underground and hands over to the generic Environmental Safety Case (ESC) following closure. However, the impact of normal operational releases of activity to the public and the environment, at all stages of the GDF lifecycle, is also considered in the generic ESC.

The effects of chemotoxic releases are not considered in this report but are addressed in the generic ESC [3].

Non-radiological environmental, social and economic effects that may arise at a generic level from implementing different GDF concepts in different geological settings (host rocks) are not considered here but will be considered as part of our Strategic Environmental Assessment (SEA) process and, as part of the planning application process, in more detail in an Environmental Impact Assessment (EIA). Our approach to sustainability appraisal and environmental assessment for geological disposal is described in [4].

Security issues associated with the operation of a GDF are not considered here.

## 1.2 Generic OSC hierarchy

### 1.2.1 Generic DSSC documents

The generic OSC is one of a series of documents that will be produced to support our regulatory submissions and to support the dialogue with our stakeholders during the development of a GDF. We use the overarching term Disposal System Safety Case to cover the hierarchy of documentation that provides the evidence to demonstrate that suitable and sufficient arrangements will be in place to ensure the safety of a GDF. The main documentation is summarised in Figure 1. At the top of the hierarchy is the DSSC Overview Report [5] (Tier 0). Below the Overview Report are three main reports written to support regulatory submissions for the three main regulatory safety themes of transport safety, operational safety, and environmental safety (Tier 1). The generic OSC deals with the safety of construction and operation of a GDF. It covers the safety regulations and requirements associated with the operation of a nuclear licensed site, referencing supporting documents where necessary.

The Tier 1 safety cases are supported by Safety Assessments, Status Reports and other supporting reports, as illustrated in Figure 1. Underpinning the Operational Safety Case is the Generic Operational Safety Assessment (Tier 2). This comprises four volumes, namely:

- Volume 1 – Construction and Non-Radiological Safety Assessment [6]
- Volume 2 – Normal Operations Operator Dose Assessment [7]
- Volume 3 – Accident Safety Assessment [8]
- Volume 4 – Criticality Safety Assessment [9]

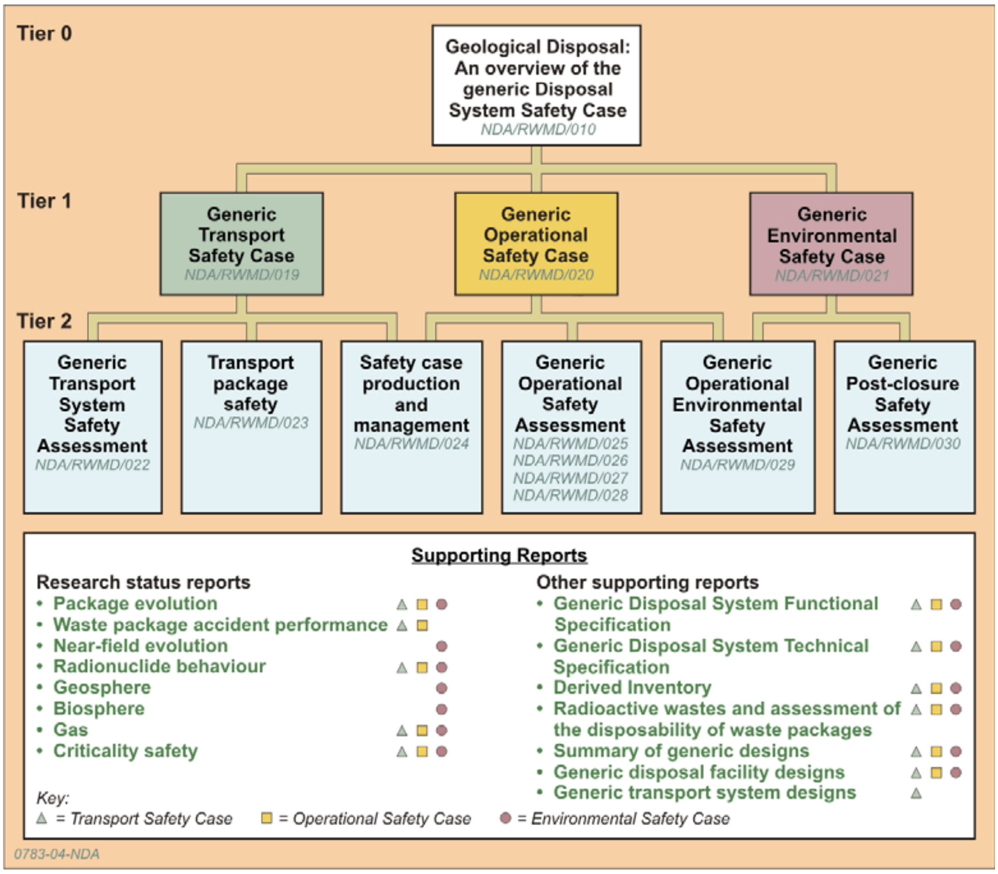
The arrangements and methodologies for the production of the generic OSC are described in the Safety Case Production and Management Report [10].

Beneath the Tier 2 reports, there are a number of supporting reports. These comprise research Status Reports and other supporting documents. The Status Reports are of two types in that they either discuss the behaviour and evolution of the engineered and natural systems or they deal with specific issues of interest to safety. The Status Reports which are of particular relevance to operational safety are those dealing with criticality safety and waste package accident performance and those which consider radionuclide behaviour and package evolution.

Other key supporting documents are the Radioactive Wastes and Assessment of the Disposability of Waste Packages [28], the Disposal System Technical Specification [18] and the GDF Generic Disposal Facility Designs Report [33].

At this stage, it is not appropriate, or possible, to produce a full set of the Tier 2 underpinning reports that would be required to support a full OSC. For this generic OSC, we have focused on developing the reports that are relevant to this generic stage. We expect both the Tier 1 and Tier 2 supporting reports to evolve significantly when we have site-specific information and designs. Below the Status Reports, there is a large body of documents, including our R&D reports, contractor and third-party reports, and scientific literature.

## Figure 1 Generic DSSC documentation hierarchy




### 1.2.2 Structure of generic OSC main report

In the next section of this report, Section 2, we provide more details on the context and objectives for the generic OSC. Section 3 sets out our safety strategy and outlines the techniques that we employ to demonstrate the safety of GDF operations.

The illustrative concept examples we have developed for different geologies prior to the selection of a specific site are described in Section 4. An overview of the radioactive wastes destined for the GDF is also presented, together with the number and types of waste packages that are expected during the operational period. We explain how the safety attributes afforded by the conditioning of the radioactive waste, the waste containers and the transport containers contribute to operational safety.

In Section 5 we outline the safety assessment methodologies we have developed and illustrate our approach with reference to an example assessment. We summarise the findings of each of the four assessment reports [6, 7, 8 and 9] and discuss the means by which we aim to ensure that hazards will be controlled such that risks to the public and workers are acceptable.

In Section 6, we summarise our key findings and outline where further work is planned to address remaining areas of uncertainty.

## 2 Context and objectives

The context for the generic OSC relates to the current stage in the MRWS site selection process and explains in more detail why the safety case is being produced (its objectives), when, why and how it will be updated with time, and the framework against which it will be evaluated.

### 2.1 MRWS – a step-wise process

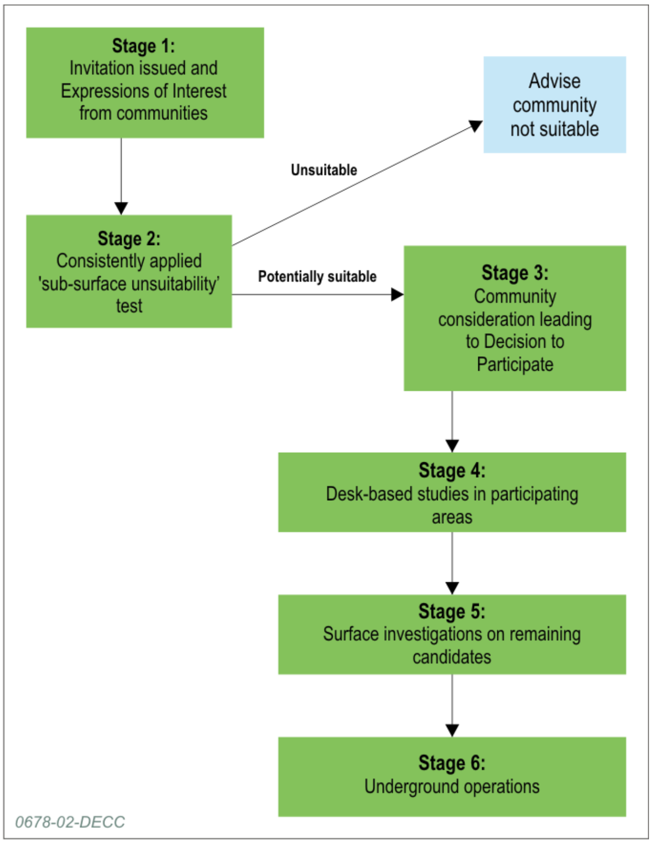
Within the MRWS programme, the UK Government has set out in a White Paper [1], the framework for the implementation of geological disposal in England, Wales, and Northern Ireland. RWMD is responsible overall for the development of a GDF, and has prepared a document [11] describing steps towards implementation of geological disposal on the basis of the framework set by the UK Government in the White Paper. That document sets out in generic terms the plans for establishing and operating a GDF, and further site-specific plans will be developed as the decision process for the siting of a GDF develops.

Whichever site is eventually chosen, the basic philosophy and methodology to ensure safe operations will be the same, although some changes to the safety arguments are expected to be made as the GDF design is developed. Although it is not possible at this stage to describe site-specific operational issues, the present generic OSC sets out the principal means by which safe operations at the selected GDF location will be ensured.

The UK Government framework for implementation of geological disposal sets out a series of stages as a means of siting a GDF, and identifies the decisions to be made and the stakeholders involved at each stage. Figure 2 shows the stages in this process.

Further information regarding the subsequent stages of the proposed programme of work is provided in the Steps Towards Implementation report [11].

**Figure 2 Stages in MRWS site selection**




## 2.2 Regulatory context

The safety of nuclear installations in the UK is assured by a system of regulatory control based on a licensing process whereby a corporate body is granted a licence to use a site for specified activities. Such activities include the processing and long-term storage of radioactive waste. The GDF will require a licence under the Nuclear Installations Act 1965, and the Government recognises that this may require legislative changes. Work is in hand to develop the required legislation which is anticipated to be in the form of a modification to the Nuclear Installations Regulations 1971.

The licensing authority for nuclear installations is the Health and Safety Executive (HSE). This licensing function is administered by the HSE's Nuclear Directorate (ND) with the day-to-day exercise of that function being delegated to the Nuclear Installations Inspectorate (NII) within the ND.

The Nuclear Installation Act places the responsibility for the safety of a nuclear installation on the licensee. Before granting a licence therefore, the HSE must be satisfied that the applicant will have an adequate management structure, capability and resources to discharge the obligations and liabilities connected with holding that licence. In accordance with these requirements [12] we have developed a Safety and Environmental Management Prospectus [13], in order to demonstrate that RWMD has an adequate management organisation, safety management arrangements, capability and resources to undertake the nuclear safety and environmental aspects of its current and near future stages of work. The prospectus will, in due course, underpin our safety case and will be maintained and updated in line with the development of a GDF. Further details regarding the purpose and content of the prospectus are described in our Safety Case Production and Management Report [10].

The Nuclear Installations Act requires the HSE to attach conditions to each nuclear site licence that it believes are necessary or desirable in the interests of safety or with respect to the handling, treatment and disposal of nuclear materials (including radioactive wastes). HSE has developed a standard set of 36 licence conditions, which are attached to all nuclear site licences and form the basis for regulation of the site by the HSE through the NII.

Of these, a number contain requirements relating to safety cases. One of these, Licence Condition 14, requires the licensee to establish arrangements for the preparation and assessment of safety cases to justify safety during design, construction, manufacture, commissioning, operation and decommissioning of facilities on the site. The actual requirement to demonstrate the safety of operations via a safety case then arises from Licence Condition 23. Further licence conditions then extend the requirement for justification via safety cases to all phases of the life cycle of a nuclear installation; these are Licence Condition 19 (Construction and Installation), Licence Condition 21 (Commissioning), Licence Condition 20 and Licence Condition 22 (Modifications) and Licence Condition 35 (Decommissioning).

The licence conditions do not specify the contents of the safety case or the nature of the arrangements for its production beyond the requirement for “adequate documentation to substantiate the safety of the proposals”. However, the regulator has produced a set of Safety Assessment Principles [20] which provide guidance on the production of safety cases and the limits and targets that the safety cases should address. In addition the regulator has produced a Technical Assessment Guide [14] which provides more specific guidance on the content of safety cases. The regulator expects the organisation’s arrangements for the production of safety cases to be in place prior to granting a site licence to an applicant and will expect to scrutinise the developing design safety case produced via these arrangements as part of its consideration of a site licence application. The generic OSC is the first stage in the development of such a safety case.

Other licence conditions which will be of particular importance to a GDF are 4, 32 and 34 which relate to the acceptance, accumulation and containment of radioactive material and radioactive waste. With the exception of disposal, the HSE regulates all aspects of radioactive waste management on nuclear licensed sites. The environment agencies are responsible for regulating the disposal of radioactive waste under the Environmental Permitting Regulations 2010. In regulating nuclear licensed sites and as a statutory requirement, HSE and the environment agencies consult one another. The arrangements are set out in memoranda of understanding which cover the regulation of nuclear safety and radioactive waste management on nuclear licensed sites and the disposal or discharge of radioactive waste on or from such sites. For any new development where a nuclear site licence is required, the HSE and the environment agencies will work together to ensure that common hold points are agreed with the developer early in the development programme [15].

A GDF will also be subject to the general provisions of the Health and Safety at Work Act (1974). These general requirements are complemented by the Ionising Radiations Regulations 1999 (IRRs) which provide for protection of workers and other persons in all industries which involve work with ionising radiations.

In addition to the requirements of the nuclear site licence, any application relating to the disposal of radioactive wastes at a GDF in England or Wales would need to satisfy the Environmental Permitting Regulations (EPR 2010), and would be supported by the ESC. EPR 2010 largely supersede the Radioactive Substances Act 1993 (RSA 93), but much of what was contained in RSA 93 now appears in Schedule 23 of EPR 2010. The EA’s expectations with regard to the environmental safety case are set out in its Guidance for Requirements for Authorisation. Our arrangements for ensuring that we meet the requirements of the EPR 2010 and the EA's guidance during the operational phase of a GDF are described in the generic ESC.

Other relevant legislation applicable to the construction and operation of a GDF is described in Appendix A of this report.

## 2.3 Objectives of the generic OSC

The set of reports (including this report) that comprise the generic OSC present the key safety arguments concerning the operational safety of a GDF and are substantiated by a structured collection of supporting analyses and evidence. The OSC is being developed in an iterative manner, in order to support the development of our programme and to promote dialogue with stakeholders.

At the current stage in the MRWS process, prior to the identification of potential sites, we cannot produce an OSC using site-specific information. The location and its geology will determine key characteristics of the GDF such as the dimensions of the underground disposal vaults where the waste packages will be emplaced and the means of underground access. Such features could have a major impact on waste package handling and emplacement operations.

In developing the generic DSSC, we have reported our work on operational safety in the form of a generic OSC in order to:

- Present safety information in a way that informs the regulators and the wider stakeholder community in a clear and concise manner to facilitate their engagement in the developing safety arguments;
- Provide a foundation for desk-based studies under the MRWS site assessment process;
- Provide the basis for the safety case documentation that, in due course, will be required to meet regulatory Site Licence Conditions;
- Provide the basis for the Letter of Compliance disposability assessment process;
- Support the scoping of the impacts of a GDF;
- Support the development of the disposal system specification, engineering designs and safety case methodology; and
- Support prioritisation of the R&D programme.

This document, the generic OSC main report, has been structured as a top-tier operational safety report. It is intended:

- To present the results of our operational safety assessment work;
- To provide assurance that, in due course, we will be able to make a safety case for a GDF that will satisfy all regulatory requirements;
- To explain how we will protect operators and members of the public from hazards arising during normal and fault conditions;
- To present the safety assessment methodology we have employed to identify and analyse potential hazards during normal operations and under fault conditions;
- To provide clear references to the documents containing more detailed information;
- To identify our forward programme of work for developing and improving our OSC.

The generic OSC will provide the basis for further operational safety case development work, initially desk-based studies, once potential candidate sites have been identified through the MRWS site selection process. As the programme evolves the OSC will also be used to update the basis for waste package disposability assessments and advice to waste producers and to inform choices between GDF design options.

# 3 Safety strategy

RWMD has defined safety, environmental, security and safeguards principles for the design process [16]. These define the objectives of geological disposal as being to ensure that all disposals of solid radioactive waste are made in a way that protects the people and the environment, now and in the future, commands public confidence and is cost-effective. In order to protect all groups of people who could potentially suffer harm as a result of GDF operations, we have defined a set of overall operational safety objectives for the operational phase of the GDF life cycle [10]. These objectives are in line with the fundamental principles that guide the design process [16].

These objectives are:

1. To ensure that waste packages accepted into the GDF meet all relevant radiological criteria[3] to ensure that exposures to the workforce and public can be restricted so far as is reasonably practicable;
2. To ensure the integrity of waste package containment and shielding under normal operating conditions and to prevent or restrict releases of radioactivity or external exposure to radiation under foreseeable fault conditions from their receipt onto site to their emplacement in underground vaults;
3. To eliminate the potential for criticality involving fissile material within waste packages;
4. To protect workers and members of the public from radiation hazards associated with waste packages;
5. To ensure the continuing integrity of waste package containment and shielding during the period from emplacement to backfilling;
6. To ensure that the methods and materials employed for backfilling and final closure of the facility do not compromise the containment of waste packages at the time of backfilling and closure nor undermine the multi-barrier concept for retention of radioactive species over the very long term post-closure;
7. To ensure that construction and waste package handling operations are segregated such that neither generates hazards affecting the safety of the other;
8. To ensure that all hazards associated with construction and excavation activities are adequately controlled.

Our safety strategy is focused on achieving these objectives.

This section sets out our safety strategy in terms of: our design strategy; our assessment strategy and our management strategy. Further information can be found in our Safety Case Production and Management Report [10].

[3] Before a GDF is operational, waste acceptance criteria (WAC) would be defined for waste packages and would have the effect of setting limits on external dose rates and non-fixed surface contamination. Exposures to the workforce would be restricted primarily by ensuring that the relevant criteria are met. We have not yet defined the WAC for a UK GDF but it is expected that these will evolve from the existing Generic Waste Package Specification, which is discussed in Section 4.2.2.

## 3.1 Design strategy

### 3.1.1 Design requirements

Where direct discharge to the environment is unacceptable, concentrating and containing radioactive waste and isolating it from the environment is the internationally accepted strategy for the long-term safe management of such materials. The development of a GDF will follow internationally accepted practices in the use of multiple barriers to achieve safety through defence in depth.

The specification, design and construction of the GDF are fundamental in protecting the workplace and the public. We are therefore putting in place processes that will ensure that the final design of the GDF will reduce risks as low as reasonably practicable, with the emphasis on the elimination of faults by design and the prevention of fault sequence development in preference to mitigation wherever possible.

We have developed a Disposal System Specification (DSS) which sets out what is required of the GDF and provides the basis for the engineering designs and the safety and environmental assessments that will underpin the generic DSSC. As such, the DSS is central to the design and assessment work. It comprises two documents:

- the Disposal System Functional Specification (DSFS) which describes the basic generic requirements and conditions applicable to the disposal system and is in a form suitable for a wide range of stakeholders [17]; and
- the Disposal System Technical Specification (DSTS) which describes in more detail the requirements and constraints on the disposal system, together with a justification for each requirement [18].

The DSFS provides the high-level specification for the disposal system and defines the nature and characteristics of the waste packages to be accepted.

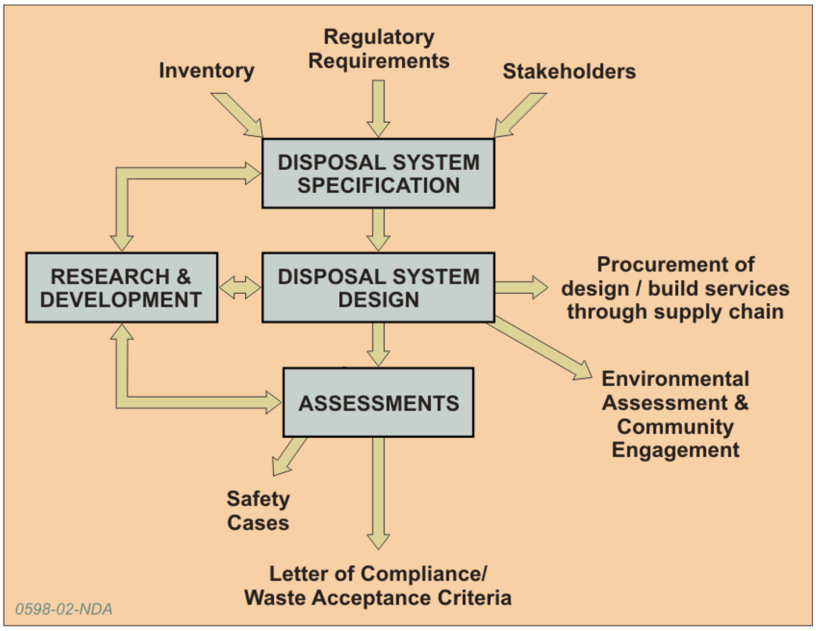
The DSTS develops this high-level specification in more detail and provides the designers of the disposal system with the requirements that must be satisfied by the design. The requirements are presented as a series of statements, which identify the requirements that the disposal system must meet. Each statement is supported by further information providing the justification for the requirement.

These documents currently describe generic requirements, reflecting the fact that a site and a disposal concept have yet to be selected. They will be periodically updated throughout implementation of the GDF, for example to respond to changes in regulations and to respond to learning from undertaking assessments. The DSTS in particular will evolve from generic to site-specific requirements as site-specific information becomes available.

Updating the DSS will take into account the results from work on the inventory, engineering design, site investigations, safety, environmental and sustainability assessment, consideration of security and safeguards issues, research and development and public, stakeholder and regulatory engagement. Figure 3 gives an overview of the iterative process by which the DSS incorporates external sources of information to guide the design and assessment processes, which in turn leads to refinements and changes in the DSS.

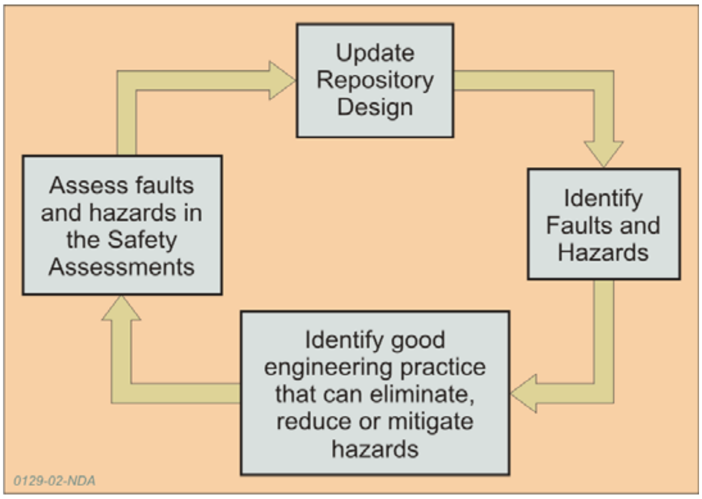
Since Figure 3 represents a high-level illustration of the process, there are inevitably a number of intermediate steps in the process which are not shown. For example, regulatory requirements and expectations, some of which are applicable to a wide range of nuclear plants, require some interpretation to adapt them for GDF use. The development of GDF specific requirements is not shown in Figure 3. Similarly the inner loop of design and safety iteration shown in Figure 4 is not shown in Figure 3. Note also that as a GDF design is developed we will undertake design option studies for key elements of the design to provide a robust justification for the final form of the design. This is not shown in Figure 3 but will be embedded in our design process. The broader optioneering process will be informed, in part, by safety considerations. Our approach to optioneering is described in [19].

**Figure 3 Iterative disposal system development**




To achieve the optimum level of safety requires close interaction between the processes relating to the specification of the requirements of the disposal system, the development of a facility design and the safety assessment of the design. Development of the design of a GDF will therefore be an iterative process supported, and in part informed by, input from the safety and environmental assessments, as shown in Figure 4 below.

**Figure 4 Iterative design process**




The iterative loop shown in Figure 4 will continue as development of the design and safety case continues. Regulatory hold points throughout the facility's life will require regulators to assess the safety case and agree or give consent to commence the next stage (for example; construction, commissioning, operation, decommissioning, closure). Safety Functional Requirements (SFRs) will be used to provide the formal, auditable link between the safety assessment work and the design. SFRs are discussed in Section 3.1.3 below.

### 3.1.2 Design safety principles

We have developed a set of principles for safety, environmental protection, security and safeguards which should be applied from the earliest stages of the design process [16]. Their purpose is to provide a basis for the design and assessment of geological disposal facilities and associated transport systems. The objective is to build into the different elements of the design, safety features which complement each other and contribute to overall safety by, for example, providing several lines of defence against the realisation of potential hazards.

The principles have been developed to conform to relevant good practice and to reflect the expectations of the regulators. They are drawn from high level international guidance on the safety of radioactive waste management in general and have been benchmarked against the NII Safety Assessment Principles [20]; the IAEA Fundamental Safety Principles [21]; the IAEA safety requirements for the Geological Disposal of Radioactive Waste [22] and the Environment Agency's Guidance on Requirements for Authorisation [15]. The principles are part of the process of safety-assured design, described in our Safety Case Production and Management Report [10].

The set of principles described in [16] are high-level principles which will be applied across all aspects of GDF design. The fundamental principles in [16] have been applied in formulating our overall safety objectives [10] for operations both on the surface and underground as outlined at the beginning of this section. Examples of the design measures and process/procedural measures by which we aim to achieve our safety objectives are described below.

**Objective 1:** To ensure that waste packages accepted into the GDF meet all relevant radiological criteria to ensure that exposures to the workforce and public can be restricted so far as is reasonably practicable;

- Specification by RWMD of wasteform content and composition, external dose rate and contamination limits (consistent with regulatory requirements) for acceptance of waste packages and transport containers supported by audit of waste packagers' package production and monitoring arrangements;
- Receipt monitoring of all transport packages for radiation and contamination will be carried out prior to transfer underground.

**Objective 2:** To ensure the integrity of waste package containment and shielding under normal operating conditions and to prevent or restrict releases of radioactivity or external exposure to radiation under foreseeable fault conditions from their receipt onto site to their emplacement in underground vaults;

- Specification by RWMD of:
  - Robust designs for waste packages and their lifting and stacking, features;
  - Impact performance criteria for all waste package designs such that releases in impact events will be minimised;
  - Fire performance criteria for waste packages such that releases of radionuclides and other hazardous materials are low and predictable
- Higher inventory waste packages will be handled in robust transport containers with very high impact resistance during surface receipt and transfer underground and will only be removed from these within shielded containment cells;
- The frequency of impact events involving waste packages will be restricted by:
  - The design of road and rail layouts for transfer of packages, parking facilities and vehicular access to buildings;

  - The restriction, by engineered and administrative means, of the speed of vehicles and lifting equipment for the transfer of packages at the surface, in a drift or a shaft and underground;
  - The avoidance by design of the need to lift packages and the minimisation of lift heights and avoidance of lifting packages over one another where lifting is unavoidable;
  - The inclusion of intrinsic, fail-safe systems on lifting systems to prevent dropped load or overspeed lowering events;
- The potential for significant fires that might challenge package containment will be minimised by:
  - Minimisation of flammable inventory in all facility areas;
  - Segregation of the package transfer routes from any significant flammable inventory;

**<u>Objective 3</u>:** To eliminate the potential for criticality involving fissile material within waste packages;

- RWMD:
  - Specify safe fissile masses for each waste package for each waste stream, as part of the Letter of Compliance process;
  - Require waste packagers to produce Criticality Compliance Assurance Documentation for every packaged waste stream;
  - Audit the waste packagers' systems and procedures for the control of the fissile material content of waste packages;

**<u>Objective 4</u>:** To protect workers and members of the public from radiation hazards associated with waste packages;

- Specification by RWMD of wasteform content and composition, and external dose rate limits (consistent with regulatory requirements) for acceptance of packages supported by audit of waste packagers' package production and monitoring arrangements;
- Location of site fences to ensure adequate distance between waste packages on the surface and potential points of approach by members of the public;
- Removal of higher inventory packages from their heavily shielded transport containers only within shielded containment cells;
- Provision of interlock systems on access doors to shielded containments.

**<u>Objective 5</u>:** To ensure the continuing integrity of waste package containment and shielding during the period from emplacement to backfilling;

- Specification by RWMD of:
  - A target period over which waste containers are designed to maintain their integrity;
  - Properties of the wasteform taking due account of potential age-related degradation phenomena;
  - Limits on corrosion promoting chemical contaminants on package surfaces;
  - Minimum load tolerance criteria for stacking for each package type;

- Optimisation of environmental conditions in vaults through conditioning of the storage atmosphere and control of water ingress to minimise the potential for loss of package integrity through corrosion.

**Objective 6:** To ensure that the methods and materials employed for backfilling and final closure of the facility do not compromise the containment of waste packages at the time of backfilling and closure nor undermine the multi-barrier concept for retention of radioactive species over the very long term post-closure;

- Backfilling and closure using only materials demonstrated through research and development programmes not to compromise package containment integrity and to provide the required levels of retention of radioactivity over the very long term;
- Backfilling and closure using techniques and equipment and at a rate that preserves the integrity of waste packages;
- Removal from the GDF, prior to backfilling and closure, of any material that may be detrimental to the performance of the engineered barrier system required for very long term retention of radioactive material.

**Objective 7:** To ensure that construction and waste package handling operations are segregated such that neither generates hazards affecting the safety of the other;

- The segregation of construction access and working areas and ventilation systems from waste package transfer routes, emplacement facilities and ventilation systems;
- Development of robust safety management systems to ensure communication and coordination between construction and waste package emplacement teams.

**Objective 8:** To ensure that all hazards associated with construction and excavation activities are adequately controlled.

- Use of proven techniques with an established safety record within relevant industries;
- Reducing the frequency and magnitude of fires and explosions by:
  - Employment of manual or automatic fire suppression systems;
  - Limitation of flammable inventories (e.g. use of low flammability hydraulic fluids);
  - Segregation of explosive stores from working areas and segregation of explosives and detonator stores;
- Controlling exposures to dusts, potentially explosive gases and noxious fumes and gases by the employment of dedicated ventilation systems meeting all relevant modern standards for underground working.

It is intended that these preliminary design aims will feed into the future development of Safety Functional Requirements as described in Section 3.1.3 below.

### 3.1.3 Safety Functional Requirements

Safety Functional Requirements (SFRs) are formal statements of the functions the design must provide to ensure that a satisfactory safety case can be made. Within the iterative design process shown in Figure 4, SFRs provide the formal auditable link between the safety analysis work that identifies faults and assesses their consequences and the design provisions that either eliminate faults or mitigate their consequences.

Safety requirements are set down in terms of design functionality so that designers have the freedom to provide the most appropriate way to implement the required functions. For example, it may be important for the safety case to ensure that the failure rate of a lifting device that results in a dropped load is less than 1 in 10,000 lifts. This is a functional requirement for the lifting device. The designer then has to select or design a lifting device that meets this requirement. SFRs are agreed between designers and safety specialists. The discussions involved in the process of reaching agreement are a valuable part of the process of ensuring a safety assured design. Design substantiation documentation provides, amongst other things, a demonstration of how the SFRs have been addressed. The safety case can then confidently claim the functionality required of the design in the fault assessment process.

It is not possible to develop a complete set of SFRs for a GDF at this stage since our designs are illustrative and not fully detailed. However, work on a preliminary set of SFRs that capture key safety case requirements on the design is already underway.

Early definition of SFRs helps ensure that our designers are made aware of safety requirements as the design evolves. To facilitate this, we define SFRs at different levels of detail, namely [10]:

- **Level 1:** Identifies the principal safety functions the design must provide and are closely linked to the Design Safety Principles described in Section 3.1.2
- **Level 2:** Sets the requirements for the safety measures required to achieve the safety functions
- **Level 3:** Specifies specific performance and design criteria for the safety measures identified.

Using this framework we can progressively refine the safety requirements as the design develops through systematic safety analysis to identify the necessary safety functions. In this way we ensure that all potential hazards are adequately controlled.

### 3.1.4 Concept development

The approach we take, until such time as site-specific information becomes available, is to define three generic geological environments, encompassing typical UK geologies. Illustrative geological disposal concept examples have been developed for each of these generic environments, in order to carry out the engineering design and safety assessments that underpin the generic DSSC. These examples are built on extensive R&D carried out by us and our “sister” organisations overseas. However, this does not mean that any of the concept examples developed will necessarily be the concept(s) used in the geological environment of a selected site.

As these concept examples are purely illustrative in nature, it is not appropriate at this stage to carry out detailed design option studies. However, pertinent findings from the example safety assessments will be fed through to the next design stage where appropriate.

When we start to develop site-specific designs in support of Stages 4-6 of the site selection process, we will utilise the iterative process described in Section 3.1.1 where the assessments influence the design to ensure that option selection is informed by technical, safety and environmental considerations. Design changes to the GDF are managed through an internal RWMD process [23] that evaluates the significance of each change and the implications that proposed changes might have on a wide range of issues including safety, environmental and security aspects.

The illustrative geological disposal concept examples chosen for the generic GDF are based on three different host rock types, namely:

- higher strength rocks (igneous; metamorphic; sedimentary) e.g. granite;
- lower strength sedimentary rocks;
- evaporites.

An overview of each of these concept examples is provided in Section 4.

## 3.2 Safety assessment strategy

### 3.2.1 Introduction

Safety assessment essentially involves the examination of the safety of a facility under both normal operating and fault conditions. Work with ionising radiations has an associated risk even when a facility is operating as intended. It is therefore important to:

- identify all potential sources of exposure under normal operating conditions;
- assess the doses likely to be incurred;
- identify and assess the measures in place to restrict those doses;
- identify any additional measures required to restrict exposures As Low As Reasonably Practicable (ALARP)

Fault analysis involves the identification of all the potential faults associated with a nuclear plant and the demonstration that they are adequately controlled. There are two key aspects to the assessment process. They are:

- the evaluation of consequences, frequencies and risks in order to ensure that they meet the relevant safety targets;
- the demonstration that the engineered design is to a high standard and robust against potential faults such that the associated risks are restricted to levels that are ALARP.

The requirement to demonstrate that risks are ALARP for both normal operating and fault conditions is a legal requirement and a key component of any safety case.

We have developed a staged approach to the development of the safety case. At this early stage it is not possible or desirable to produce a comprehensive assessment. Rather we are developing our processes and methodologies for the production of safety cases and have carried out example assessments as an illustration of our approach. This has been done in order to identify additional data and development work required to allow us to progress to the next stage of project implementation. It is important to note that, in the assessment presented in the following sections, we have followed the generally accepted approach of allowing for uncertainties by the inclusion of conservatisms in our assumptions. It is expected that the level of conservatism in our assessments will be progressively reduced during the development of a site-specific design.

As part of the generic DSSC, we have produced a Safety Case Production and Management Report [10], which describes the management arrangements, regulatory constraints, safety principles and procedures and methodologies used for the development of the generic OSC and TSC, in order to provide confidence that the safety cases for transport and operation are as robust as is possible at this early stage in the development of the Disposal System concept.

### 3.2.2 Safety documentation strategy

As described in Section 2, application for, and maintenance of, a nuclear site licence will require, among other things, the submission to HSE of documentary safety justifications for the construction, operation and closure of the facility to be built on the site. These submissions provide the written justification that the requirements described in Section 3.2.1 above have been met, namely: that risks have been reduced to a level that is ALARP and that the relevant safety criteria have been met.

We will develop a suite of safety documentation to justify the safety of a GDF as it is developed from preliminary site-specific designs, through the design process, to an operational facility. We have proposed a strategy for development of the safety case for a GDF, based on relevant good practice and HSE guidance [14]. The strategy is based on a phased approach to safety documentation reflecting the development of the project from the current generic concept through to implementation and operation.

In the lifecycle of any nuclear facility there are a number of key phases which will require specific consideration. These are:

- Site investigation and design;
- construction and installation;
- commissioning (inactive and active);
- pre-operation;
- operation;
- decommissioning and closure.

Appropriate safety documentation will be provided in line with major programme stages: identification of potential candidate sites, site characterisation, planning approval, facility construction, facility operation, facility sealing and closure, and eventual withdrawal of control. Safety cases will also be required for decommissioning of redundant facilities, likely to take place immediately prior to and during the closure phase. The period from the start of the building of the geological disposal facility until closure, may extend up to 100 years or even more.

In conjunction with the regulators, we are developing a Permissions Schedule for the implementation of geological disposal, which sets out the permissions to be obtained at the various stages of the MRWS site selection process and covers all regulatory disciplines. Once emplacement has commenced, the safety case documentation will require periodic review and will need to be updated in line with facility modifications such as the construction of additional vaults. Following emplacement, additional safety documentation will require to be produced and approved to allow backfilling and closure operations to proceed. Decommissioning activities will require to be supported by a decommissioning safety case.

When complete, the Permissions Schedule for Geological Disposal will set out:

- the regulatory submissions to be produced;
- the assessments to be made by regulatory organisations;
- the permissions which will be needed at various stages in the site selection process set out in the UK Government's MRWS White Paper [1].

At each stage the safety case is required to demonstrate, to an appropriate level of detail, that:

- the proposed installation has been soundly assessed and meets the required safety principles and criteria;
- the installation conforms to good nuclear engineering practice and to the appropriate standards and codes of practice;
- the installation will be adequately safe during normal operations and under fault conditions. The analysis should identify the safety measures that need to be implemented to realise the required safety standards;
- the installation is, and will remain, fit for purpose;

- the level of radiological risk to the workforce and members of the public is as low as reasonably practicable. This demonstration should include the options that have been considered and justify those chosen;
- the installation has a defined and acceptable safe operating envelope, with defined limits and conditions.

The Safety Case Production and Management Report [10] describes in more detail the safety documentation required at each stage in the development of a GDF.

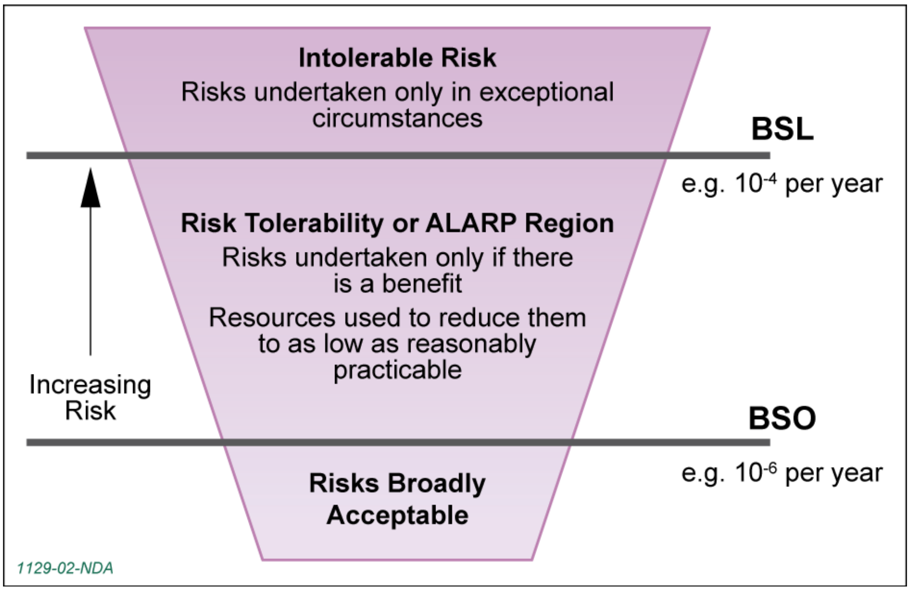
### 3.2.3 Assessment criteria

The nuclear industry uses the concept of risk[4] to define acceptability criteria against which the results / conclusions of safety analyses may be judged. HSE thinking on risk is presented in their publications Tolerability of Risk from Nuclear Power Plants [24] and Reducing Risks, Protecting People [25] and is best illustrated by the Tolerability of Risk (TOR) framework shown in Figure 5. HSE believe there is a certain level of risk that may be said to be unacceptable or 'intolerable' irrespective of the level of benefit accruing from the activity generating the risk except in exceptional circumstances. This is represented by the top region in Figure 5. The bottom region represents the level of risk deemed to be 'Broadly Acceptable'. Risks in this region are generally regarded as being insignificant and adequately controlled but still need to be shown to be ALARP. Finally, risks in the intermediate region are regarded as 'Tolerable' providing they have been appropriately assessed and it can be demonstrated that they are being maintained as low as reasonably practicable (i.e. ALARP).

Numerical targets (risk or dose) representing the boundary between the intolerable and tolerable regions are defined as Basic Safety Levels (BSLs) in the SAPs. New nuclear facilities are expected, as a minimum, to be compliant with the BSLs. Some BSLs are legal limits under the IRRs.

Numerical targets (risks or doses) representing the level at which risks become broadly acceptable are defined as Basic Safety Objectives (BSOs). BSOs are regarded as benchmarks that reflect modern nuclear safety standards and expectations.

**Figure 5 Tolerability of risk framework**




[4] Some criteria are expressed directly in terms of risk. Others are expressed as dose limits but they are derived from a consideration of risk.

In addition to the above, is the key principle that runs through all safety justifications, namely that when the risk has been shown to fall between the BSL and BSO, a justification must be provided to demonstrate that there is nothing more that could reasonably be done to reduce the risk still further. Even where the BSO has been met, licensees are still obliged to reduce risks where this is reasonably practicable.

Our Radiological Protection Policy Manual [26], describes the acceptability criteria we will use to judge the results of our analyses. Specific criteria are also presented in Section 5 of this document where these are relevant to the assessment.

## 3.3 Management approach

It is recognised that the technical requirements for a robust safety case are unlikely to be met, unless the organisation responsible for delivering safety has an appropriate management system in place and adequate capabilities and resources to meet its commitments.

A Safety and Environmental Management Prospectus [13] has been developed that details our safety management system and demonstrates our commitment to health and safety. Fuller details of our safety management arrangements are provided in the Safety Case Production and Management Report [10]. In that report we describe the safety management systems which will govern the development of the safety case for constructing and operating a GDF. The report also provides a clear statement about the company, its structure and how we propose to operate.

In addition, a Management Systems Manual has been compiled which covers management systems such as document control, document review, programme and project management, many of which also support the safety and environmental management systems.

In parallel with the Management Systems Manual, we are also producing a Safety Case Manual, which will contain the detailed procedures and methodologies we are developing to support the preparation of safety cases for the GDF, as outlined in the Safety Case Production and Management Report.

# 4 Assessment basis

## 4.1 Introduction

Geological disposal involves constructing an engineered facility, between 200 and 1,000 metres underground, in which radioactive wastes can be placed. Internationally, one thing all geological disposal systems have in common is the use of a multi-barrier system to isolate the wastes from the environment and ensure the radioactivity in the wastes is sufficiently contained so that it will not be released back to the surface in unacceptable amounts that may cause harm to people and the environment.

A number of multi-barrier disposal concepts have been developed internationally, each driven by different requirements (for example, by the amount and type of waste, or societal preferences) and geological host setting. These can be used to provide a basis for generic geological disposal concepts that could be adapted to provide safe and secure geological disposal of the higher-activity wastes and materials arising in the UK.

The geological environments available for a GDF will depend on the locations of sites identified through the MRWS selection process. As different disposal concepts are suited to different geological environments, it is not possible to select any single disposal concept at this time. However, in order to develop our approach we have defined three generic geological environments, encompassing typical UK geologies. For each of the generic environments defined, illustrative concept examples have been developed on which to carry out example engineering design and safety assessments for the generic DSSC. These illustrative concept examples are described later in the Section. It should be noted these examples may be used to support ongoing engagement through the MRWS process although this does not mean that any of these concept examples will necessarily be the concept used for any selected site.

The following sections describe the nature and quantity of the wastes destined for a GDF; describe the characteristics of the barriers provided by the wasteforms and waste containers, and present an overview of the illustrative concept examples developed for the different geologies.

## 4.2 Waste, wasteforms and packages

### 4.2.1 Waste inventory

For waste management purposes radioactive waste is divided into a number of waste categories defined in relation to their nature and radioactive content.

These generic categories are defined formally in the Glossary and are described below:

- **Low-level waste (LLW)**. LLW comprises building rubble, soil and steel items such as framework, pipework and reinforcement from the dismantling and demolition of nuclear reactors and other nuclear facilities and the clean-up of nuclear sites. However, currently most LLW is from the operation of nuclear facilities, and this is mainly paper, plastics and scrap metal items. Only a small fraction of the LLW produced in the UK will be consigned to a GDF.
- **Intermediate-level waste (ILW)**. ILW exists in a wide range of physical and chemical forms including metals, graphite, concrete and other rubble, sludges, flocs and various organic materials including oils. Its major components are metal items such as nuclear fuel casing and nuclear reactor components, graphite from reactor cores, and sludges from the treatment of radioactive liquid effluents.

- **High-level waste (HLW)**. HLW is initially produced as a concentrated nitric acid solution containing fission products from the primary stage of reprocessing SF. HLW is currently incorporated into borosilicate glass (vitrified) in stainless steel canisters at the Sellafield Waste Vitrification Plant (WVP). It is expected that, by 2015 most of the HLW that is expected to arise will have been treated in this way. HLW generates significant amounts of heat until such time as the short-lived fission products have decayed. Vitrified HLW will be stored for at least 50 years to allow a significant proportion of the radionuclides to undergo significant radioactive decay such that they are more suitable for disposal. Further packaging of the canisters is likely to be required to create a waste package suitable for transport and disposal.

In addition to these wastes, the UK Government has specified other radioactive materials that could possibly come to be regarded as waste in the future and so are included in the Baseline Inventory for disposal. These materials are:

- **Spent nuclear fuel (SF)**. Most of the SF from the UK's existing civil reactors has been, or will be, reprocessed to separate plutonium and uranium which could be used to make new fuel and so will not be consigned to a GDF. However, some SF from UK advanced gas-cooled reactor (AGR) power stations and all SF from the Sizewell B pressurised water reactor (PWR) may not be reprocessed and so would be managed as waste. A new build programme would generate further arisings of SF.
- **Plutonium (Pu)**. The UK stocks of separated plutonium are not currently declared as waste but are held because they may have a future use, for example in the manufacture of some reactor fuels. If they were however to be disposed of in a GDF, they would first need to be converted into a suitable stable wasteform. For the purposes of the generic DSSC, we have assumed that the separated plutonium is converted into a wasteform that allows it to be disposed of using the same disposal concept as for HLW and SF.
- **Uranium (U)**. Stocks of uranium come from refining uranium ore to make fuel or from reprocessing SF. These stocks include some enriched uranium which is suitable for making fuel for reactors, but the majority of the inventory by volume comprises depleted, natural and low-enriched uranium (DNLEU) residues ('tails') from fuel production. For the purposes of the generic DSSC, we have assumed that the DNLEU is converted into a stable wasteform that allows it to be disposed of using the same disposal concept as the LLW and ILW, and that the small quantity of high-enriched uranium (HEU) is converted into a wasteform that allows it to be disposed of using the HLW/SF disposal concept.

In conjunction with the Department for Energy and Climate Change, we maintain a compilation of data on UK radioactive waste holdings – the UK Radioactive Waste Inventory (UK RWI) [27]. Information from the 2007 UK RWI was used to compile the 'Baseline Inventory' of radioactive waste and materials for geological disposal, given in the UK Government's White Paper [1].

However, the Baseline Inventory does not contain sufficient detail to provide a basis for the designs and assessments underpinning the generic DSSC. We have therefore developed derived inventories for each group of materials included in the Baseline Inventory. Building on the UK RWI, the Derived Inventory is compiled from projections made by the Site Licence Companies on the basis of their current assumptions regarding the nature, scale and timing of future operations and activities that will result in the generation and conditioning of radioactive waste.

The DSTS [18] makes use of the derived inventory reports to define our Derived Inventory Reference Case which is consistent with the Baseline Inventory. The derived inventory provides a waste stream by waste stream breakdown of the radioactive and physical/chemical inventory on a facility and per package basis. Summary information is provided in Table 1 – Table 3.

The DSTS also defines an Upper Inventory. The Upper Inventory considers changes to the Baseline inventory that might arise for example from the operation and decommissioning of new nuclear power stations and defines an increased volume of conditioned waste that could require geological disposal. The implications of the Upper Inventory for the fault analysis work in the generic OSC are discussed in section 5.9.

Further details on the quantities and characteristics of the wastes in the inventory, including Upper Inventory estimates, are provided in our Radioactive Wastes and Assessment of the Disposability of Waste Packages Report [28] and the Disposal System Technical Specification [18].

The example assessment presented in the generic OSC focuses on the radiological inventory. Consideration is also given to other, non-radioactive but potentially toxic materials which may be present in the waste. The nature of such materials varies considerably between individual waste streams but they tend to be present in relatively small quantities. Generally, it is found that the engineered safety barriers and other safety systems designed to prevent the release of and potential exposure to radioactive materials provide suitable protection against other toxic materials. This aspect will be considered in more detail as the design is developed. Protection against exposure to hazardous materials during construction activities is discussed in Section 5.12.

**Table 1 Derived inventory reference case for higher activity wastes**

| Materials | | Activity at 2040 (TBq) | Packaged volume ($m^3$) |
|---|---|---|---|
| HLW | | $1.7 \times 10^7$ | 7,457 |
| ILW | UILW | $1.6 \times 10^6$ | 361,692 |
| | SILW | $1.5 \times 10^4$ | |
| LLW | | $8.2 \times 10^{-2}$ | 16,632 |
| SF | | $3.8 \times 10^7$ | 10,363 |
| Plutonium | | $3.8 \times 10^6$ | 6,989 |
| Uranium (DNLEU & HEU) | | $5.0 \times 10^3$ | 94,502 |

### 4.2.2 Wasteforms and containers

The physical containment and form of the radioactive waste will play a significant role in the safety arguments for operations at a GDF.

#### ILW and LLW packages

Standards and specifications for the packaging of ILW and LLW wastes have been developed, in order to provide confidence that waste packages currently being manufactured, will be compatible with the safe operation of the GDF. Packaging requirements are detailed in the Generic Waste Package Specification (GWPS) and supporting documents [29].

In conjunction with the waste producers, we have specified standard features for waste containers that are suitable for packaging the vast majority of ILW and LLW being put forward for disposal at a GDF. Figure 6 illustrates a range of packages that have been developed.

- 500 litre drum;
- 3 cubic metre box (3m$^3$ box);
- 3 cubic metre drum (3m$^3$ drum);
- 4 metre box (4m box);
- 2 metre box (2m box);
- MGBWS box;
- WAGR box.

The four waste containers shown in the top half of Figure 6 are unshielded intermediate level waste (UILW) containers. They have no internal shielding and are transported in a reusable transport container (the container and contents then forming a Type B Transport Package). The three waste containers shown in the lower half of Figure 6 are shielded intermediate level waste (SILW) or low level waste (LLW) containers. Packages using these containers either have in-built shielding or contain low activity materials and thus may be handled by conventional techniques (i.e. remote handling is not essential). In most cases shielded waste packages are also designed to qualify as transport packages in their own right.

Waste producers develop their own containers and these must meet the requirements set out in the GWPS.

It is a requirement that all wastes for disposal at a GDF must be in a stable solid form in order to restrict the release of activity in the event of a loss of containment fault and to minimise post-closure transport through the environment [28]. In many cases (although not universally) this requires the wastes to be immobilised into a solid matrix or wasteform. The nature of the wasteform will depend on the physical, chemical and radiological properties of the waste. For ILW and LLW a number of processes have been developed and shown to be suitable for the conditioning of a wide range of wastes, for example:

- Immobilisation of particulate or slurry waste by in-drum mixing with a liquid grout (either cementitious or polymer-based) which hardens to form a monolithic wasteform;
- Backfilling of drums or boxes containing heterogeneous solid wastes with a cementitious grout;
- Supercompaction of soft wastes, such as plastics, in sacrificial drums, followed by placement into a container and backfilling with a cementitious grout.

**Figure 6 Waste containers for ILW and LLW**

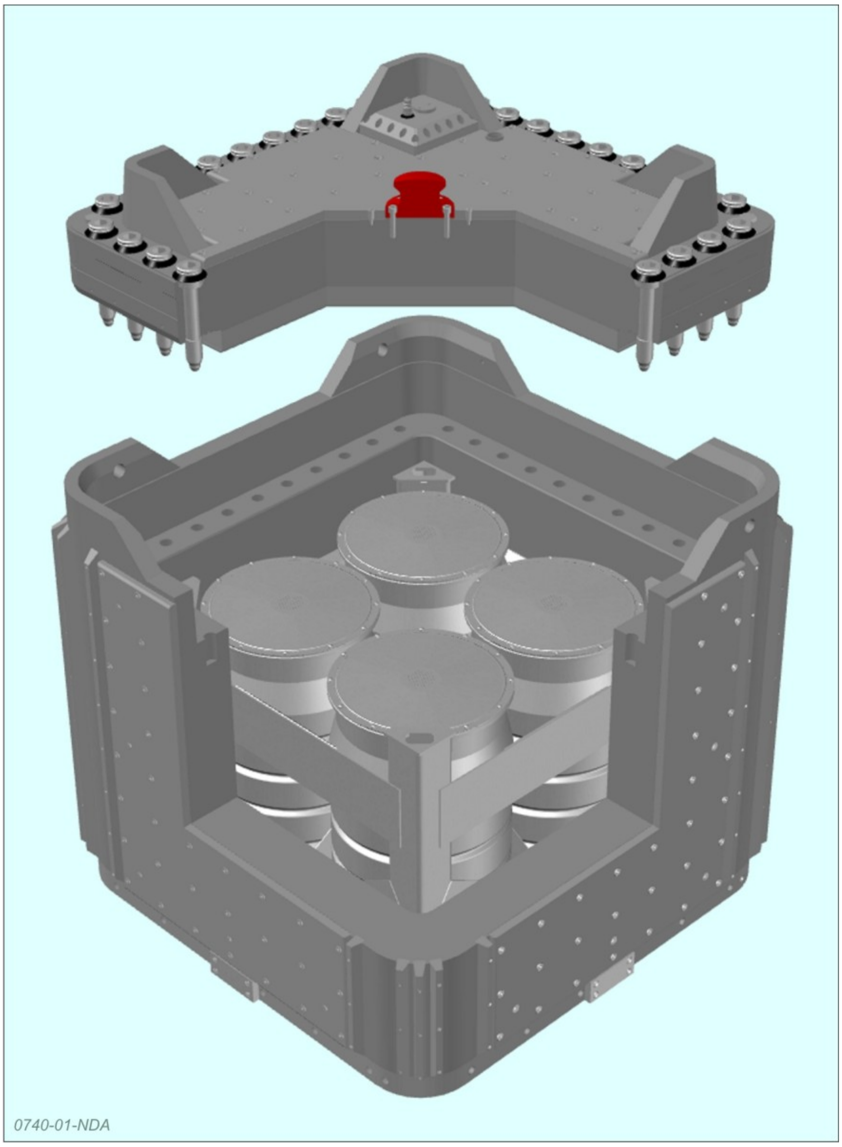

Figure 7 shows a cutaway view of immobilised waste in a 500 litre drum. For discrete items that may contain internal voids that cannot be infiltrated by an encapsulant to form a completely solid matrix, the waste may additionally be surrounded by an annular liner, base and cap composed of inactive cementitious material.

**Figure 7 Cutaway of 500 litre drum showing drum and waste**


The 500 litre drums, $3m^3$ boxes and $3m^3$ drums are unshielded waste packages that require remote handling. During transport and surface handling operations, these waste packages will be held in shielded transport containers (see Figure 8). A conceptual design has been developed for a Standard Waste Transport Container (SWTC). The SWTC would incorporate either 285mm or 70mm of steel shielding. Each SWTC would hold either one $3m^3$ box or one $3m^3$ drum or four 500 litre drums in a stillage. The combination of SWTC and waste package(s) would meet the requirements for a Type B(U) transport package as defined in the IAEA transport regulations [30]. Type B transport packages must meet stringent impact and fire resistance criteria. The waste packages will only be removed from the SWTCs once the loaded transport container has been transferred underground at the GDF to the remote handling cell line. The empty transport containers will be returned to the surface for subsequent re-use.

Some wastes will be packaged in stainless steel boxes, 2 m or 4 m in length, which may be concrete lined, providing integral shielding. These packages are designed to be transported and handled without any additional shielding or containment and will be compliant with the IAEA definition and performance requirements for an Industrial Package (IP). In line with the IAEA Transport Regulations, the allowable contents of these packages are limited to wastes that qualify as Low Specific Activity materials and/or Surface Contaminated Objects. Due to the low inventory, IP2s require substantially less fire and impact resistance than Type B transport packages.

**Figure 8 Cutaway of a Standard Waste Transport Container (SWTC)**




**High level waste and spent fuel packages**

In the illustrative concept examples (discussed in Section 4.3) the HLW and SF will be packaged in robust disposal canisters, manufactured either from copper with cast iron internal furniture or from carbon steel (copper and steel variants would have similar dimensions and robustness). Although a specification for HLW and SF has been developed [31], final decisions on the design of canister and construction materials have not yet been made as this will depend upon the host geological environment and detailed GDF design adopted for a particular site. The disposal canisters manufactured from copper are illustrated in Figure 9.

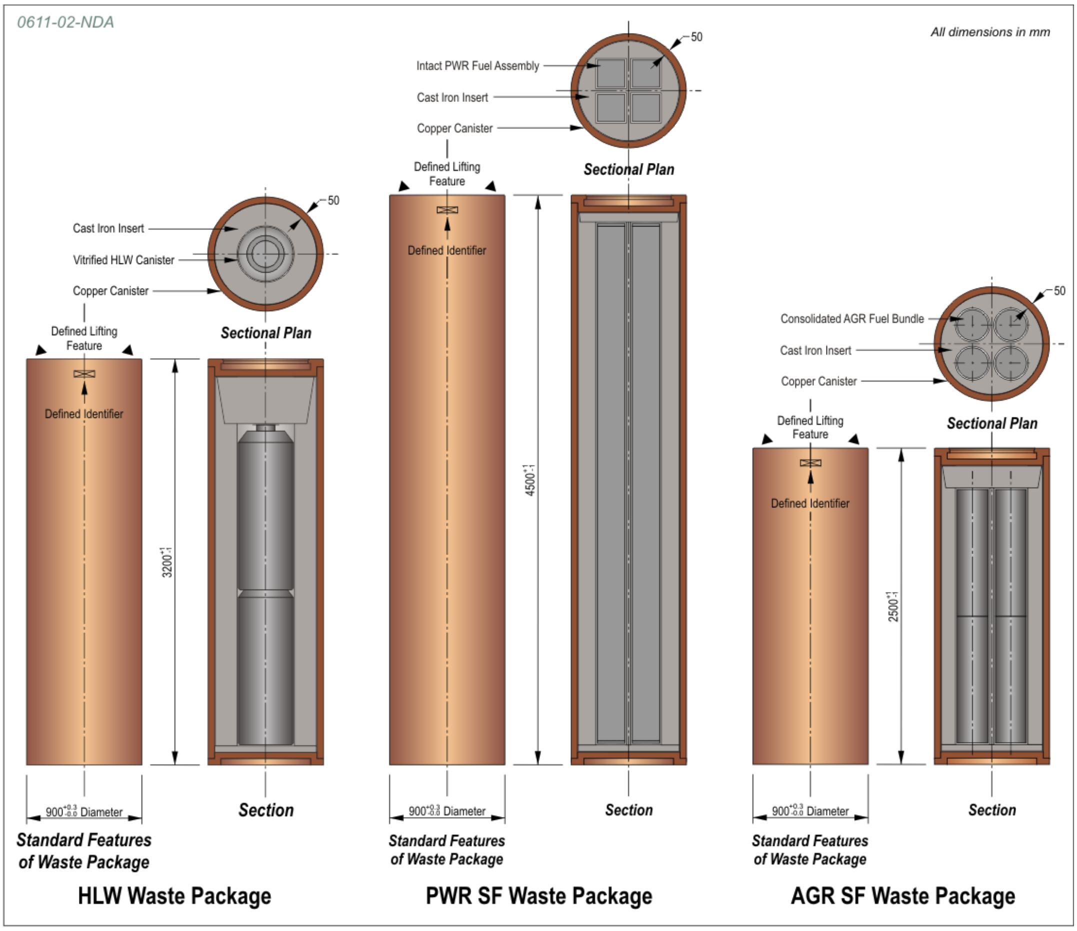
HLW comprises high active liquors immobilised in a vitrified wasteform inside stainless steel canisters known as standard waste vitrification plant (WVP) canisters. There would be two VWP canisters stacked within each disposal canister, located and shielded by a concentric cast iron insert as shown in Figure 9.

At present, it is anticipated that PWR spent fuel would be packaged as complete fuel assemblies while AGR spent fuel would be packaged as consolidated bundles of fuel pins in a stainless steel basket. PWR and AGR fuel is manufactured in a ceramic form.

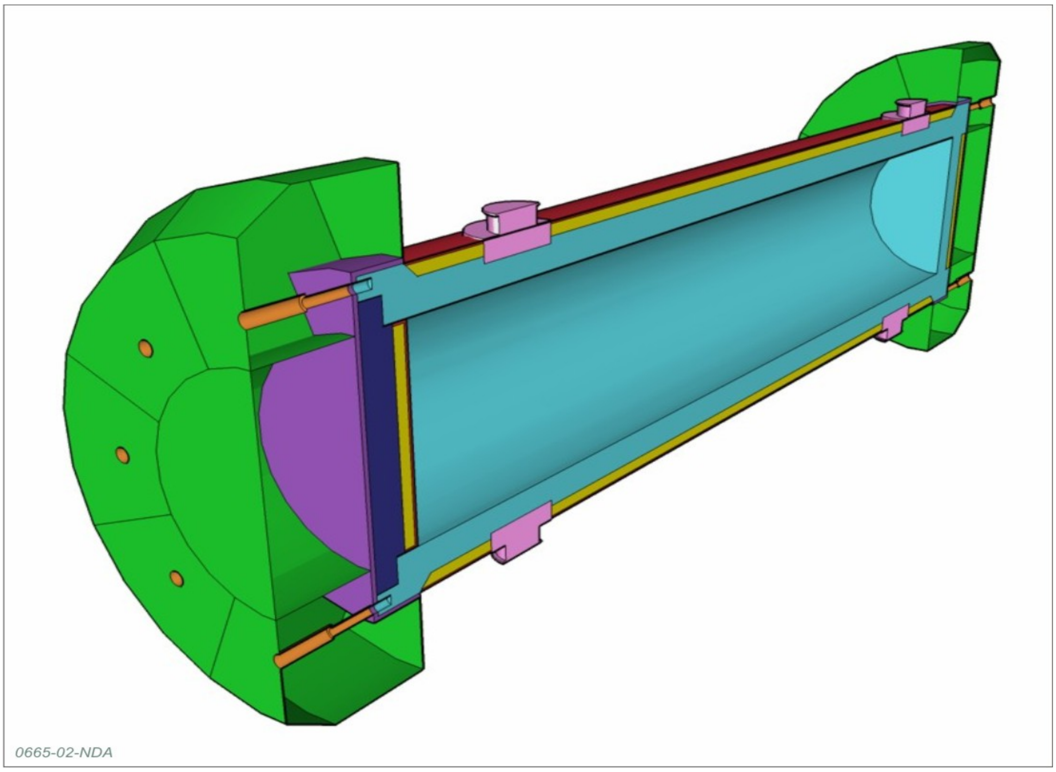
Ceramics and vitrified wasteforms generally perform well in terms of long-term containment, and provide a passive wasteform for impact and fire faults.

Because of their very high radioactive inventories, these packages would need to be transported to the GDF in shielded flasks. We have called these disposal canister transport containers or DCTCs. They would comply with the requirements for a Type B transport package under IAEA Transport Regulations for a concept developed for such DCTCs (see Figure 10).

## Figure 9 Waste containers for spent fuel and HLW



## Figure 10 Cutaway of computer generated view of DCTC




### Plutonium/uranium packages

At this stage it has been assumed that plutonium and high enriched uranium (HEU), if declared as waste, would be packaged in a similar design of disposal canister to that proposed for HLW/SF and would also be transported to a GDF in a DCTC. The nature of the packaging process has yet to be determined but, in order to enable safety assessments to be conducted for the generic DSSC, it is assumed that Pu/HEU would be converted into titanium-based ceramic pucks, with multiple pucks being placed in a stainless steel can. Multiple cans of pucks would then be encapsulated in glass in a large canister which would, in turn, be placed in the disposal canister.

It is further assumed that depleted, natural and low enriched uranium would be compacted in the form of a uranium oxide into 500 litre drums. The drums would be transported in SWTCs (see Figure 8).

### Waste package numbers

The number of waste packages and transport containers expected to arrive at a GDF has been estimated for planning purposes. The following tables (Table 2 and Table 3), from [28] itemise the number of waste packages assumed for the Derived Inventory Reference Case. For simplicity, waste package numbers have been expressed in terms of disposal units where a disposal unit comprises:

- one $3m^3$ box or $3m^3$ drum
- four 500 litre drums in a stillage
- one 2 metre box or 4 metre box
- one disposal canister.

### Table 2 Projected number of ILW/LLW/DNLEU disposal units (Derived Inventory Reference Case)

| **Waste package type – ILW/LLW/DNLEU** | **Total number of disposal units** |
|---|---|
| ILW/LLW – UNSHIELDED | |
| 3m³ boxes and drums | 44,208 |
| 4 * 500 litre drum (ILW/LLW) | 39,380 |
| 4 * 500 litre drum (DNLEU) | 41,332 |
| **Total Unshielded ILW (UILW)** | **124,920** |
| | |
| **Total Shielded ILW (SILW)** | **6,310** |
| | |
| **Total LLW packages** (2metre/4metre boxes) | **892** |
| | |
| **TOTAL ILW/LLW PACKAGES** | **132,122** |

### Table 3 Projected number of HLW/SF, Pu/HEU disposal units (Derived Inventory Reference Case)

| **Waste package type – disposal canisters** | **Total number of disposal units** |
|---|---|
| HLW/SF | |
| HLW | 3,656 |
| AGR SF | 5,341 |
| PWR SF | 655 |
| **Total HLW/SF canisters** | **9,652** |
| | |
| Pu/HEU | |
| Plutonium | 3,426 |
| HEU | 50 |
| **Total Pu/HEU canisters** | **3,476** |
| | |
| **TOTAL DISPOSAL CANISTERS** | **13,128** |

The number of waste packages and transport containers expected to be received per year will determine transport logistics and the type of infrastructure needed at a GDF from receipt of packages through transfer to underground reception areas and through to emplacement of waste packages. It will also influence the size of the transport fleet required, covering such issues as the number of reusable transport containers required and transport routes to a GDF. We anticipate emplacing around 1,800 intermediate and low level waste packages per year for the first 30 years of operation decreasing thereafter. The disposal rate for HLW and SF is expected to be around 200 canisters per year. Package throughput is discussed in more detail in [33].

At this stage, we envisage that the waste packages associated with the Upper Inventory would require a GDF to operate for a longer period of time. It is recognised that the scheduling of wastes to a GDF, including any wastes represented in the Upper Inventory, will need to be considered in the context of optimisation.

All of these factors will have a bearing on the overall safety of operations at, and en route to, a GDF and will have to be taken into account in any safety assessments carried out: separate assessments are being carried out and reported in the safety case for transport to a GDF [2].

## 4.3 Overview of illustrative GDF designs

For the generic DSSC, six illustrative geological disposal concept examples have been developed in order to begin to examine the safety aspects of a GDF with respect to normal operations and potential fault conditions. For each of the three host rock types there is one concept example for ILW/LLW/DNLEU waste packages that are emplaced in vaults and another concept example for HLW/PU/SF/HEU waste canisters that are emplaced individually in vertical deposition boreholes or in horizontal deposition tunnels. The concept examples are based on geological disposal concepts that have previously been developed in the UK or are being developed under national programmes in other countries. The six concept examples are shown in Table 4 together with the published concepts on which they are based.

In the safety assessment presented here we assume that ILW/LLW/DNLEU and HLW/PU/SF/HEU illustrative concepts are co-located. By this we mean that they are disposed of within the same GDF (in separate disposal areas [33]) rather than in separate disposal facilities. This is in line with Government policy [1] which states that "In principle the UK Government sees no case for having separate facilities if one facility can be developed to provide suitable, safe containment for the Baseline Inventory."

The example concepts are described in the following sub-sections.

### Table 4 Illustrative geological disposal concepts

| Host rock | Illustrative geological disposal concept examples (developer, country) | |
|---|---|---|
| | ILW/LLW/DNLEU | HLW/SF/Pu/HEU |
| Higher strength rocks[(1)] (igneous/metamorphic/sedimentary) | UK ILW/LLW Concept (NDA, UK) | KBS-3V Concept (SKB, Sweden) |
| Lower strength sedimentary rock[(2)] | Opalinus Clay Concept (Nagra, Switzerland) | Opalinus Clay Concept (Nagra, Switzerland) |
| Evaporites[(3)] | WIPP Bedded Salt Concept (US DOE, United States) | Gorleben Salt Dome Concept (DBE Germany) |

Notes:

1. Higher strength rocks – the UK ILW/LLW concept and SKB's KBS-3V disposal concept were selected because of the availability of information on these concepts for the UK context. Supercontainer options (such as the KBS-3H concept) would also need to be considered in our future work, if a candidate community comes forward in an area of the UK providing access to suitable higher strength rock at GDF depth.
2. Lower strength sedimentary host rocks – Nagra's concepts for Opalinus Clay were selected because a recent NEA review regarded the Nagra work as state of the art [32]. However, it should be noted that there is similarly extensive information available for the French (Andra) concepts (for Callovo-Oxfordian Clay), which have also been accorded strong endorsement from international peer review. Although the Swiss concepts are used as the basis of the illustrative geological disposal concept examples in the generic ESC, as we move forward we would also draw on information from the French programme and from the Belgian HLW/SF supercontainer concept, if a candidate community comes forward in an area of the UK providing access to suitable lower strength sedimentary rock at GDF depth.

3. Evaporites – the concept developed by the US DOE for transuranic wastes at the WIPP was selected because of the wealth of information available from a licensed facility that has been operating for more than 10 years, and the concept developed by the German Company for the Construction and Operation of Waste Repositories (DBE) for HLW/SF was selected because of the level of information available for it.

### 4.3.1 Higher strength host rock

The illustrative concepts for higher strength rock are described in more detail in [33]. They are based on co-location of facilities for ILW/LLW/DNLEU and HLW/SF/PU/HEU. The concept examples comprise a number of basic elements including:

- Surface Facilities;
- Underground Access;
- ILW/LLW/DNLEU Disposal Area;
- HLW/SF/PU/HEU Disposal Area.

**Surface facilities**

The surface facilities would be divided into two separate areas. One area would be reserved solely for construction activities and the other would accommodate package receipt. As there will be a need for continuing construction work after the commencement of emplacement operations, the aim would be to separate construction and emplacement activities so far as possible with separate access and ventilation arrangements.

A typical surface layout, taken from [33], is shown in Figure C1 (Appendix C).

The construction area facilities would be the first to be developed and would be used to service underground development. Once the facility was operational, the construction area would continue to be used to provide underground access for all construction personnel and materials and the export route for excavated rock.

The main function of the package receipt area facilities would be the receipt and handling of in-coming waste packages and their transfer to the drift train for transfer to the underground emplacement facilities through the drift – an inclined tunnel.

Waste packages would be delivered to a GDF by road or rail. (It is possible that part of the package transport route would be by sea but it is expected that in this case the final stage of transport would be by road or rail.) Rail sidings and HGV parking would be provided on-site to accommodate rail wagons and lorry trailers holding transport packages awaiting transfer underground.

All unshielded waste packages and disposal canisters would be transferred underground in their transport containers. Package handling activities at the surface would therefore be limited to the transfer of transport packages from the delivery unit (lorry trailer or shunter wagon) to the drift wagon. It is anticipated there would also be facilities at the surface for the decontamination, inspection and maintenance of empty transport containers. This would include remote handling facilities to allow minor repairs to be undertaken on any incoming transport containers failing acceptance tests.

Additional active facilities providing support to the main operational plants would include liquid effluent treatment and active ventilation plants, an active laundry and laboratories. The level of radioactive hazard in such areas should be very low. There would also be facilities for the maintenance of the drift locomotive and wagons and the shunting locomotive which would be used to transfer the mainline rail wagons onto site. These facilities would not normally present any radiological hazards.

### Underground access

Underground access would be provided by a drift and shafts. The drift would be used for the transfer of waste packages underground and would be equipped with a rack-and-pinion rail system. Underground emplacement operators would also travel via the drift, but separately from the waste packages. Access for construction workers and equipment would be by vertical access shaft located in the construction area. A separate shaft would be provided for the export of excavated rock also located in the construction area. The shafts and drift would provide the ventilation intake and return routes. The underground construction and emplacement areas would have independent ventilation systems.

### UILW and ILW/LLW/DNLEU disposal areas

Unshielded package(s) would be removed from the SWTC by remote handling within an Inlet Cell. This would be a series of linked shielded containments. An overhead crane would be used to unbolt and remove the lid from the transport container, then remove the waste package and move it along the cell line for monitoring and transfer to a bogie to be taken to the disposal vaults. The exact nature of the monitoring activities has yet to be determined but would include a further check of the package identification and monitoring for surface contamination. On arrival in a disposal vault, the package would be lifted by overhead crane and placed on an emplacement stack.

The shielded waste packages have low surface dose rates and do not require remote handling. They would be emplaced in separate disposal vaults of similar dimensions. The shielded packages would be accumulated in a buffer store located near the entrance to the vault and transferred to the disposal vault in campaigns using a large capacity stacker truck.

### HLW/SF/PU/HEU disposal area

The HLW/SF/PU/HEU disposal area would consist of disposal tunnels designed for in-floor emplacement of individual disposal canisters within vertical deposition holes.

Prior to transfer from the main-line wagon to the drift locomotive (or shaft transfer system in the evaporite rock illustrative design) DCTCs would be monitored and inspected. Each HLW/SF/PU/HEU disposal canister would then be taken underground in its DCTC. In a Transfer Hall at the base of the drift, the DCTC would be turned to the vertical orientation and lowered into a pit in the floor, using an overhead crane. A heavily shielded deposition machine would be positioned above the DCTC. The disposal canister would then be lifted from the DCTC into a shielded tube within the deposition machine. The disposal canister would be transferred in the deposition machine to the disposal tunnel and positioned above the deposition hole. The deposition machine would then be rotated to the vertical and the disposal canister lowered into the hole which would be pre-lined with pre-compacted blocks and rings of bentonite clay. Further bentonite blocks would be installed above the canister using the deposition machine.

A schematic underground layout for this concept example is presented in Figure C2 (Appendix C).

## 4.3.2 Lower strength sedimentary host rock

The illustrative concept examples for lower strength sedimentary rock are described in more detail in [33]. They have many similarities to the higher strength rock concept example although the design of underground openings would be different due to the different support capabilities. In higher strength rock, it is possible to excavate stable tunnels and vaults, with spans of order 20 metres at typical GDF depths, with limited rock support. In lower strength sedimentary rock, it is possible to excavate tunnels of 10-15 metres diameter at GDF depths, but in this case the excavations are likely to require considerable engineered support. This gives rise to the main differences between the example concepts, as outlined below.

**Surface facilities and underground receipt**

Surface facilities, underground access and reception areas for the receipt of ILW/LLW/DNLEU packages would be the same as those for the higher strength rock concept except that the overall height of the reception area would be lower in some parts. Apart from potentially lower lift heights in some areas, package handling arrangements would be essentially unchanged.

The main difference anticipated is that, while for the higher strength rock, the inlet cell and disposal vaults could remain operational until closure; with the lower strength sedimentary rock, the operational life of the inlet cell and vault structures is assumed to be limited. This is because of the need for maintenance of the additional engineering support required for excavations in the lower strength sedimentary rock. The design therefore provides for a separate inlet cell for each disposal module.

**Emplacement**

The dimensions of the disposal vaults are currently assumed to be smaller than those for the higher strength rock concept. Thus, the number of vaults required would be greater.

Given the reduced vault heights, package stack heights would also be lower than in the higher strength rock concept example.

In the lower strength sedimentary rock concept, there would be no requirement to lift unshielded packages into the disposal vaults. Rather, they would be transferred horizontally through an interlocking system of shield doors.

HLW/SF/PU/HEU disposal canisters would be emplaced in disposal tunnels similar to those envisaged for the higher strength rock concept example. However, the canisters would be placed end to end horizontally in the tunnels rather than in individual deposition holes. The entrance to each tunnel would be fitted with a double set of shielded doors. The DCTC containing the disposal canister would enter the disposal tunnel reception area and the outer shield doors would be closed. Using remote means, the disposal canister would then be transferred horizontally onto a disposal trolley, preloaded with bentonite blocks. The inner shield doors would then be opened to allow the disposal trolley into the disposal tunnel. Canister and blocks would be lowered as a unit to the floor of the deposition tunnel and released from the trolley such that the canister would be supported by the bentonite blocks. The area surrounding the newly emplaced canister would then be filled with bentonite pellets using a mobile hopper.

An example underground layout [33] is depicted schematically in Figure C3 (Appendix C).

### 4.3.3 Evaporites

Details of the illustrative concept examples for evaporites, adapted for UK conditions, are provided in [33]. For these examples a bedded halite (rock salt) has been assumed; that is, the host rock is assumed to comprise a bed of salt deposit overlain by sedimentary rock. It has been further assumed that the host rock is sufficiently extensive vertically and horizontally to accommodate a GDF.

**Surface facilities and underground receipt**

The evaporites concept examples, as is currently envisaged, would not include drift access. Rather, a fourth shaft would be provided for the transfer underground of waste packages and emplacement operations personnel. HLW/SF/PU/HEU disposal canisters in DCTCs, unshielded packages in SWTCs and shielded packages would all be transferred underground in this way. It is noted that the choice of a fourth shaft is a feature of the US and German designs on which the concept example is based. Evaporite rocks exhibit 'creep' whereby the excavation boundary slowly moves and the roof and side walls 'flow' inwards. Rock bolts and mesh would be provided as a primary support where required but the excavation dimensions would gradually decrease. Hence, although the use of a drift within this geology would be technically feasible, it would have significantly higher maintenance requirements compared to the other geologies.

**Emplacement**

Operations underground would be similar to those for the lower strength sedimentary rock concept. While the height of most excavations would be reduced compared to the other concept examples, inlet cells of similar dimensions to that for the lower strength sedimentary rock would be provided for the removal of unshielded packages from the SWTCs.

The creep properties of the evaporite rocks would limit the operational time available. Therefore, in order to avoid repeated excavation activities in the vicinity of the primary remote handling facilities, a similar strategy would be proposed to that adopted for the lower strength sedimentary rock; namely, the provision of a separate inlet cell for each disposal module, constructed at the same time as the disposal module.

Due to the constraints imposed by the geology, unshielded packages would be emplaced using a remotely operated stacker truck rather than an overhead crane. As for the lower strength sedimentary rock, the packages would be transferred through shield doors from the transfer tunnel bogie to the stacker truck.

As for the lower strength sedimentary rock concept example, the HLW/SF/PU/HEU disposal canisters would be transferred to the disposal tunnels in the DCTCs. The emplacement of HLW/SF/PU/HEU disposal canisters in tunnels would be similar to that described for the lower strength sedimentary rock concept except that the canisters would be placed on the floor of the disposal tunnel. A mobile hopper would then be used to fill the area surrounding the newly emplaced canister with crushed rock salt (excavated during construction activities).

## 4.4 Basis for Generic Operational Safety Case

As explained in Section 2.3 the generic OSC provides the basis for the disposability assessments that underpin the Letter of Compliance (LoC) process. NDA RWMD assesses the disposability of packaged wastes during waste producers' development of conditioning and packaging processes before the waste packages are manufactured.

Through this process we use our generic designs and safety cases to check that the waste packages would be transportable and disposable. This process is important for the generic DSSC and the generic OSC in particular as it gives confidence that the safety case is taking due account of waste packages and is compatible with real waste packages as being developed by industry. The process is very important for waste producers too as it allows for early packaging of waste by giving confidence that investment in plant to recover and package waste will result in waste packages which can be transported and disposed to a future GDF. It also gives confidence that waste will not require repackaging before disposal in a GDF. The LoC process is discussed in [10] and is described in more detail in [34].

Prior to the production of the generic DSSC the LoC process was based on the UK ILW/LLW concept (PGRC)) [35] and for HLW/SF it was based on [36]. The LoC process requires quantified safety assessments in order to fully understand the issues associated with specific wastes and waste packaging proposals. To this end both the ILW and HLW/SF concept designs were supported by operational safety assessments [37,38].

To facilitate continued support for the LoC disposability assessment process it is necessary for the generic OSC to be sufficiently quantified to continue to serve the purpose outlined above. To this end the scope of the generic OSC includes a quantified example assessment based on the illustrative concept examples. To maintain a consistent thread with previous work the example assessment is based on the two illustrative concept examples for the higher strength rock geological environment.

The two illustrative concept examples for higher strength rock are considered to provide a good representative basis for quantified assessments because they include all receipt, handling, transfer and emplacement activities, and in the most-part present worst-case scenarios (cavern drop heights for instance). There are two exceptions to this:

(i) the concept example for evaporite which proposes the use of a shaft in preference to a drift for transferring waste packages underground. Clearly the use of a shaft introduces a completely new set of potentially significant impact faults, including cage falls and ‘cage absent’ faults. Justification of this aspect would require design basis accident analysis and, possibly, severe accident analysis of the detailed design to be undertaken in the future.

(ii) the concept examples for lower strength sedimentary rock and evaporite, introduce the use of additional shield doors (e.g. at the entrances to the ILW vaults and the HLW/SF disposal tunnels). This would increase the potential for external exposure faults resulting from failure of access controls.

Potential impacts of the differences between the concept examples on the safety assessment are discussed in Section 5.8

# 5 Safety assessment of a GDF

## 5.1 Introduction

In the generic OSC, the safety of all operations carried out within a GDF is considered, from construction and operation through to decommissioning and closure. Operational safety is largely concerned with the safe handling and care of all waste packages, from receipt at a GDF site and transfer underground, through to backfilling and closure operations, together with the decommissioning of facilities no longer required. This example assessment focuses on package handling and emplacement operations. Backfilling and closure operations will be considered in future assessments.

The generic OSC considers potential radiological impacts to workers and the public, from both normal day-to-day operations and from fault conditions during the operational period. This is done in order to ensure that the GDF is designed in such a way as to reduce routine exposure so far as reasonably practicable and within acceptable limits. Similarly, the design is scrutinised from the earliest stages to identify potential hazards and the faults which might cause them to be realised. Our safety specialists and designers work together during the development of the design in order to eliminate, by design, as many potential faults as possible. Where this is not possible, sufficient protection and safety systems would be incorporated in the design, to ensure that the consequences and risks associated with any fault condition are restricted so far as reasonably practicable and are within acceptable limits.

Suitable methodologies have been developed for carrying out safety assessment work which will be further refined as the design and safety work progresses. Since there is not, at present, a site-specific design on which to carry out a full fault analysis, we have chosen, by way of example, to trial the methodology using two of the illustrative concept examples. For this work, the two illustrative concept examples for higher strength rock have been chosen since this allows the use of slightly more detailed design features and assumptions developed for the previous generic ILW/LLW and HLW/SF concepts [35, 38]. Section 5.8 provides a qualitative discussion of the implications of the safety assessment for the illustrative concepts examples in different geologies.

It should be emphasised that there is not yet sufficient information to carry out a comprehensive assessment and therefore the following represents a 'broad brush' assessment of GDF safety as an illustration of the approach. It is also important to note that, in performing this assessment, we have followed the generally accepted approach of balancing uncertainties by the inclusion of conservatisms in the assumptions. We fully expect to be able to progressively reduce the level of conservatism during the development of a site-specific design.

Sections 5.2 to 5.11 below consider radiological safety in normal operation and under fault conditions for both workers and the public during the operation of a GDF. Section 5.12 considers non-radiological safety during construction both before and after active operations (i.e. waste emplacement) commence.

## 5.2 Normal operations

Normal operations at a GDF would include:

- The receipt of waste packages arriving by road and rail;
- Monitoring of received packages for contamination and radiation (Note that where waste packages are transported within transport containers it is the exterior of the transport container rather than the waste package that is monitored);

- Transfer of waste packages from delivery vehicle to an internal transport vehicle (including monitoring and inspection activities - note that where waste packages are transported within transport containers it is the exterior of the transport container rather than the waste package that is monitored);
- Transfer of waste packages underground;
- Transfer to and emplacement of waste packages in the appropriate disposal area (including, where necessary, removal of packages from their transport containers);
- Decontamination (where necessary) of transport containers for re-use;
- Inspection and maintenance activities;
- Monitoring and surveillance of emplaced packages;
- Construction operations in parallel with emplacement;
- Backfilling operations;
- Operation of the active liquid effluent system;
- Operation and maintenance of the active ventilation system.

(Note that the generic TSC 'hands over' to the generic OSC when a package is cleared for loading onto an internal transport vehicle within a GDF. For clarity, doses to workers from on-site operations (e.g. uncoupling of rail wagons, package monitoring) are reported with all other normal operation doses in the generic OSC and are also considered in the generic TSC)

These routine operations would, inevitably, give rise to worker doses from the following pathways:

- exposure to airborne activity from;
  - surface contamination on packages or transport containers;
  - gaseous releases from vented packages;
  - naturally occurring radon gas and its daughter products released from the host rock;
- direct radiation from packages.

A legal limit is set by the IRRs for annual radiation doses for all potentially exposed groups. For employees working with ionising radiation, the legal limit is 20 mSv per calendar year. We have, in addition, a BSO target value for those employees of no more than 1mSv per year [26]. However, the primary aim is to restrict doses to levels that are as low as is reasonably practicable. All employees who regularly work with ionising radiation would wear personal dosimeters in order that their actual exposure can be closely monitored and recorded in compliance with legal requirements.

For other employees on site (that is, those not working with ionising radiation), the limit, as set out in our Radiological Protection Policy Manual [26], is 2mSv per year, with a design target value of 0.1 mSv per year.

The corresponding limits for members of the public are discussed in the generic ESC [3] which presents the assessment of public exposure as a result of discharges and disposals (gaseous, liquid and solid) from a GDF during normal operations. A discussion of public dose from direct radiation is covered in volume 2 of the Operational Safety Assessment (OSA) [7] and Section 5.2.2 of this report.

Following the development of a site-specific design, detailed dose budgets would be estimated for all aspects of normal operations. At the present stage, preliminary estimates have been made of annual effective doses to workers during those operations that are considered likely to be the main contributors to any individual worker's annual dose. These are the doses that may be incurred in areas which process or store waste packages and which would be expected to have the highest background levels of radiation and contamination. At this time, no estimates have been made for other areas such as active liquid effluent or maintenance facilities which may become contaminated in the course of operation of the GDF. It is considered highly unlikely that exposures in such areas would contribute significantly to the annual dose of any individual worker. Similarly, routine maintenance tasks (e.g. work on the active ventilation system including any filtration systems) will not be assessed until our designs are developed further. Given the immobile nature of the waste we expect any doses incurred in maintenance activities will be very low and that they will not make a significant contribution to the annual dose of any individual worker. Maintenance activities will be subject to an ALARP assessment as the design is developed to ensure any doses incurred are as low as reasonably practical.

Details of the assessment of normal operations doses are provided in [7]. The following are the main work groups considered:

- Operators involved in coupling/uncoupling of rail wagons at the rail sidings;
- Operators exposed to the background radiation from transport packages in temporary storage at the rail sidings;
- Operators involved in coupling/uncoupling of road trailers at the lorry trailer park;
- Operators exposed to the background radiation from transport packages in temporary storage at the lorry trailer park;
- Inlet Cell operators potentially exposed to radiation from transport containers in buffer storage;
- Health physics operatives involved in radiation and contamination monitoring of transport packages;
- Operators involved in transporting shielded waste packages from their underground buffer store to the disposal vault.

The assessment [7] divides the GDF into areas and the annual doses to workers in these areas are calculated. The calculations take account of the tasks performed in each area, the estimated number of packages in the area and the occupancy of the area required for operators to perform the tasks. Data and methodologies from earlier safety assessments [e.g. 37,39] were employed to assess annual bounding doses for handling and monitoring operations requiring operators to work in close proximity to transport packages and for work in some other areas of the GDF.

### 5.2.1 Exposure to airborne activity

There is the potential for exposure to airborne activity from, for example, surface contamination on packages or transport containers. However, this is not anticipated to be significant due to the strict limits imposed by the IAEA Transport Regulations [30] and our package specifications (e.g. GWPS [29]) on such contamination and the requirement for monitoring prior to dispatch and on receipt.

There would also be radioactive releases from vented packages although the release of particulates (but not gases) would be restricted by vent filters required by our waste package specifications [29]. The majority of gaseous release from the waste would come from unshielded intermediate level waste (UILW), which is handled remotely, and any emissions from this source would be removed by the ventilation system. Man entry into the UILW vaults would not be permitted and so exposure to workers from this source is unlikely. By contrast man entry into the SILW vaults is a requirement of the SILW emplacement process in the current illustrative concept design and an assessment of the dose contribution from airborne activity generated by the wastes in this area has been made. HLW/SF/PU/HEU packages are not expected to be vented and would not therefore contribute to airborne activity during normal operations at a GDF.

Workers underground could potentially be exposed to naturally occurring radon gas and its daughter products. Radon levels will be dependent on the nature and characteristics of the host rock. Because of this no attempt to quantify this dose contribution has been made at this stage. However, as a site-specific design is developed, the ventilation system would be designed to adequately restrict radon concentration levels in operating areas.

Filtered ventilation systems would be provided to remove any particulate airborne activity. Appropriate monitoring of airborne activity levels, including radon, will be provided as necessary in line with the requirements of the IRRs. Given the nature of the operations to be undertaken at a GDF, the limits placed on transport package contamination and the filtration of package vents, airborne activity levels in operating areas are expected to be consistent with achieving design targets for annual dose rates. This is discussed further in [7].

Public doses from normal operational (authorised) discharges and disposals are expected to be below legal limits and consistent with modern standards of good practice. These are discussed in the Operational Environmental Safety Assessment [40].

### 5.2.2 Exposure to direct radiation

**Operator doses**

Direct radiation dose rates on the exterior of transport packages would be constrained to low levels by the IAEA Transport Regulations [30] and by our package specifications [29]. Waste package direct radiation dose rates would be measured and checked prior to dispatch and on receipt at a GDF and at various stages during emplacement operations. Within a GDF, shielded facilities will be provided where necessary to limit direct radiation doses to acceptable levels. Appropriate monitoring of direct radiation levels will be provided as necessary in line with the requirements of the IRRs.

For a number of operations carried out at the surface, some mitigatory measures were assessed to be required to be included in the developed design to maintain individual annual doses below the design target. These mitigations would include carrying out certain operations remotely (e.g. uncoupling of rail wagons). Some other operations (e.g. monitoring of transport packages) would be carried out away from areas where numbers of packages would be temporarily stored since such areas would have relatively high ambient radiation levels. In addition, access to areas where numbers of packages would be temporarily stored, such as the rail sidings and the trailer park would be restricted because of their elevated background radiation levels. With these mitigations in place, operator doses for surface operations are predicted [7] to be below the design target. Annual doses in surface areas are anticipated to be independent of host geology.

Underground, operations in one particular area – the inlet cell operating area – were assessed as requiring further mitigation to meet the design target. The dose in this area would be from transport packages containing unshielded waste packages stored in the buffer storage area. In this case, the mitigation is likely to be an increase in the shielding between the two areas.

The next most significant exposures are assessed to be those occurring during emplacement of shielded waste packages in their disposal vault. While these are not assessed to require further mitigation to meet the design target, it is possible that future assessments would find some additional mitigatory measures to be reasonably practicable to implement. An example of simple measures which could be put in place are provision of training for stacker truck drivers in an inactive area in order to reduce the time spent in close vicinity to the working face in the disposal vault. It may well be that this would be more effective than, for example, the provision of shielding on the driver's cab which could potentially restrict visibility and thereby increase exposure times and doses.

Doses incurred in underground disposal areas (rail receipt area, buffer store and disposal vault) would be affected by the host geology and the detailed design. This is because the ambient radiation levels would be dependent on the number of packages in the front rank of emplaced packages. This number would be expected to be lower for the evaporite and lower strength sedimentary rock geologies in which vault cross sections might be smaller. In these cases, doses are assessed to be significantly lower than for the higher strength rock case.

In addition to the careful design of working practices, management arrangements and procedures, routine exposure will be limited through the provision of appropriate shielding, restriction of access to high background areas and design for remote operation to the degree that is reasonably practicable. It is therefore concluded that there is confidence that the design target BSOs [26] will be met for workers who work with ionising radiations, and other on-site workers.

**Public doses**

Public doses from direct radiation during normal operations at a GDF have been assessed assuming an individual spends approximately 1 hour per day at the external site fence. The assumed source of the direct radiation dose is a train of 12 wagons, each loaded with a transport container emitting 0.1mSv per hour at 1m (the package specification limit) parked in the siding closest to the site boundary. The assessed dose is four times the design target of 0.01mSv per year [26] However, it is unlikely that a train of twelve wagons, as described above, would remain in the sidings unprocessed for a full year as a 'just in time' delivery regime is assumed in the illustrative concept. This means a train starts to be processed on arrival at a GDF and the conservative average occupancy of the rail sidings would only be six wagons. A further conservatism in our analysis is the assumption that all transport containers are at the package specification limit for external doses. In practice there would be a distribution of external doses below the maximum permitted value.

### 5.2.3 Future work

Further work will be performed on the quantification of potential doses from inhalation of airborne contamination emanating from packages and from inhalation of natural radon from the host rock.

Further work on normal operational doses to GDF operators will extend the assessment to those areas not directly involved in the processing of waste packages and to groups of operators who may spend time working in a number of different areas. This work will also consider visitors to the site and on-site workers not working with radiation (e.g. administrative staff).

## 5.3 Radiological faults

A fault condition arises when an item of plant or equipment malfunctions, when an erroneous process condition arises or when an operator forgets to do something or carries out a task incorrectly. In the safety case we need to consider what faults may occur during the lifetime of a facility and ensure that there are suitable and sufficient measures in place to prevent such faults from resulting in exposure to operators or the public, or to mitigate the consequences of any faults that do occur.

In the following sub-sections, we present the results of the trial analysis carried out based on illustrative concept examples.

### 5.3.1 Fault identification

To understand what faults can develop with any part of a plant or process, formal hazard identification studies are carried out. These studies are used extensively in many industries, as a first step in assessing what might go wrong. The hazard identification process provides the corner stone for the subsequent fault analysis.

Once a site-specific design is available, a key part of this process will be the development of comprehensive fault schedules for the operation of the GDF considering the totality of all its various component parts and their mutual interfaces. The fault schedule normally identifies:

- The initiating event or cause;
- The initiating event frequency;
- Description of fault progression;
- The consequences of the fault (who is affected and what type and magnitude of hazard they may be exposed to);
- The control measures that will either prevent the progression of the fault sequence or reduce its consequences should it progress.

At this stage, we do not have sufficiently detailed designs to develop comprehensive fault schedules. However, potential faults have been studied for the illustrative concept examples. Through this work we believe we have identified the significant types of fault that may occur during the operation of a GDF.

Hazard and operability (HAZOP) studies were carried out in the 1990s for the PGRC reference design for ILW/LLW packages [37]. These involved round-table structured and systematic examinations of the design at that stage by a multi-disciplinary team of designers and safety specialists. Separate hazard identification studies were subsequently carried out for the UK HLW/SF reference concept [41] and these are considered appropriate for our KBS-3V illustrative concept.

The fault schedules developed from these original studies have been progressively updated and extended in recent years, using a variety of formal hazard identification techniques such as review workshops; structured brain-storming sessions and review of fault schedules for similar facilities. Full details of these studies are provided in [42], which includes a full auditable trail of the process.

We have drawn on this work to produce preliminary fault schedules [42] for the generic DSSC for a co-located GDF for disposal of ILW/LLW/DNLEU and HLW/SF/PU/HEU packages on a single site. This example assessment is based on a sub-set of faults for which an initial quantitative assessment is considered viable at this early stage of design and for which it was felt it would be instructive to provide early feedback to the design team. However, the importance of preserving a complete record of all the faults identified is recognised, since these could be applicable to a site-specific design, and so all fault sets are listed in full in [42].

Given the potential requirement to include Pu, HEU and DNLEU packages in a GDF inventory, the existing fault schedules were reviewed to determine whether there were any additional faults associated with these packages. It was concluded that, due to the similarities of the waste materials and the packaging employed to those considered for the ILW/LLW and HLW/SF, the existing fault schedule was sufficiently comprehensive to cover operations involving these additional waste inventories [8] for all the illustrative concepts shown in Table 4. It was noted, however, that neutron radiation would potentially comprise a more significant contribution to external exposure from Pu/HEU packages than other waste packages, and may require the provision of appropriate neutron monitoring facilities.

With the exception of criticality faults, the schedule of faults for quantitative analysis is presented in the supporting assessment report [8]. Criticality faults, including those specific to Pu/HEU packages, are discussed in [9].

The hazard identification procedure employed for the current assessment identified a number of internal hazards (e.g. fires, rock-falls and vehicular impacts) and external hazards (e.g. aircraft impact and seismicity). However, it is recognised that, at this stage in the development of the GDF design, meaningful assessment of certain internal hazards, such as internal flooding, is precluded due to a lack of site-specific design information. Similarly, assessment of other external hazards (both man-made and natural) will require a knowledge of the specific site on which the GDF is constructed and its surroundings. In the current assessment, we have addressed those internal hazards and external hazards to the level of rigour that is practicable at this stage. A more comprehensive identification and assessment of these hazards will be possible once a site has been selected and a more detailed design is available.

The main faults identified as having the potential to give rise to a radiological hazard and which are also applicable to all our illustrative concepts, are:-

- Damage due to impact
    - *Transport containers or waste packages could be accidentally dropped from a crane during lifting; this might be due to mechanical failure of the crane or human error in the lifting process. Very rarely, shock loading to the lifting system (due, for example, to a seismic event) might dislodge lifting systems or loads. However, the designs of cranes and structures will be suitably qualified to withstand such events. Other potential impact events include vehicle impacts or a drop of one package onto another. Impact damage to a transport package or waste package could, in the worst case, result in a loss of shielding and/or release of airborne particulate radioactivity;*
- Damage due to fire
    - *If a fire occurred in any of the facilities for waste package handling or disposal, then fire damage such as seal degradation could potentially lead to a loss of containment and release of airborne radioactive material from the waste packages involved;*
- Inadequate shielding
    - *If, due to a failure at an upstream plant, a transport container or waste package did not have adequate shielding or contained excessive inventory, then radiation levels from the package would be higher than expected. If not detected, this could potentially lead to workers involved in package handling, inspection or monitoring being exposed to higher than normal dose rates;*
- Contamination
    - *Elevated levels of surface contamination on transport containers or waste packages could lead to resuspension in air of active particulate and dose to operators via inhalation;*

The assessment of criticality faults [9] has shown that the only potential challenges to criticality safety during the operational phase lie in failure on the part of waste producers to restrict package fissile material inventories to within limits derived from detailed criticality analysis.

For all other identified fault sequences (impacts, fires, floods, etc.) which could have criticality as their outcome, the LoC process and our waste package specifications ensure a suitably low fissile content for ILW/LLW packages and robust waste packages for higher fissile inventories which ensure that criticality is not a credible consequence [9].

## 5.4 Design basis accident analysis

### 5.4.1 Introduction

In order to progress the development of our methodology, we have carried out an example safety assessment for our preliminary sub-set of quantifiable faults. As described previously, this example assessment is based on illustrative concept examples for higher strength rock, with additional information drawn, as necessary, from previous work for this type of geology [33, 35].

The reason for performing such an analysis at this stage is to identify faults whose consequences are potentially so severe that the design of the facility should address them specifically, such that they are either eliminated or they are adequately protected against through engineered safety systems. The output of the analysis, which establishes whether faults within the design basis are adequately protected against, is an important input to the design process. This analysis will be repeated on an iterative basis as the design of a GDF is developed.

### 5.4.2 DBA criteria

#### Identification of design basis faults

The purpose of DBA analysis for developing designs for new plant is to assist the development of the design by identifying the design features and safety systems required for safe operation. The first step is to identify the Design Basis Faults (DBFs). These are the faults with the greatest harm potential in terms of initiating fault frequency and radiological consequences. A number of criteria are used to identify DBFs and they are set down in full in our Radiological Protection Policy Manual (RPPM) [26]. The principal criteria relate to initiating fault frequencies and fault consequences (in terms of doses to workers and the public). For convenience, the dose targets that are used as criteria for selecting DBFs are reproduced in Table 5.

**Table 5 Dose targets for design basis fault sequences**

| | Initiating Fault Frequencies (per year) | Basic Safety Level (mSv) | Basic Safety Objective (mSv) |
|---|---|---|---|
| Off-Site (Public) | $> 10^{-3}$ | 1 | 0.01 |
| | $10^{-3} – 10^{-4}$ | 10 | |
| | $< 10^{-4}$ | 100 | |
| On-Site (Worker) | $> 10^{-3}$ | 20 | 0.1 |
| | $10^{-3} – 10^{-4}$ | 200 | |
| | $< 10^{-4}$ | 500 | |

The identification of DBFs begins with a calculation of “partially protected” doses for all the potential DBFs that have been identified. Partially protected doses are calculated for each fault only taking credit for the passive protective features of the plant (e.g. structural walls that provide shielding against direct radiation). Active safety features of the plant (e.g. filtration systems) are disregarded at this stage of the analysis. The partially protected doses for both workers and the public are then compared to the Basic Safety Limits (BSLs) in the frequency band appropriate to the frequency of the initiating faults. Any fault with partially protected doses that exceed the relevant BSL for either workers or the public is designated a DBF and is subject to further analysis. Although faults that result in partially protected dose consequences below both the worker and public BSLs do not require further safety provisions at this stage because their consequences are already low, an ALARP assessment (see Section 5.10) may result in additional safety provisions to reduce the fault consequences further. Although faults that do not meet the criteria for DBFs are not subject to further design basis analysis, they may be included in the probabilistic safety assessment (PSA) – see Section 5.5.

For the DBA analysis presented here, we have taken a more cautious approach to identifying DBFs than that described above. We have disregarded the initiating fault frequencies and compared all fault consequences to the BSL values appropriate to the highest frequency bands in 45. Specifically, a fault has been designated as a DBF if, on the basis of the partially protected dose, the consequences are greater than:

- 20 mSv to workers, or
- 1 mSv to members of the public.

These doses are lower than those in the higher frequency bands shown in Table 5 and so provide a more demanding set of criteria for defining DBFs. We believe that at this stage of our work this is an appropriate approach as initiating fault frequencies for our illustrative concept designs cannot be well defined at this stage. Our conservative approach ensures that faults are not omitted from the design basis due to incorrect frequency assignment.

**Adequacy of safety measures**

Having identified the design basis faults, the DBA analysis then proceeds with the examination of the set of DBFs to determine whether any might be eliminated by modification to the design or, where this is not practicable, to demonstrate that the safety systems provided to protect against each fault are suitable and sufficient as judged against criteria defined in our Radiological Protection Policy Manual (RPPM).

This is done by calculating the “protected doses” for both workers and the public and comparing them to the criteria in Table 5. The protected dose calculation takes account of both passive and active safety provisions and will therefore be lower than the partially protected dose used to identify DBFs. As for the identification of DBFs, we take the more cautious approach of using the most restrictive BSLs (i.e. those for the highest frequency band) to judge the acceptability of the safety provisions.

The RPPM [26] also stipulates specific requirements regarding the number and nature of safety systems required to protect against DBFs.

Specifically, for DBFs, the single failure criterion must be met. That is, the ability to deliver the required safety function should not be removed by any single random failure within the systems provided to deliver that function or their supporting systems. This leads to the basic requirement for at least two independent safety systems to protect against any DBF sequence such that, if one system has an unrevealed failure, the remaining system is suitable and sufficient by itself to provide the required degree of protection. It is emphasised that these are minimum requirements. Additional systems could still be included in the design to the extent which is reasonably practicable. Indeed the aim is to comply with the single failure criterion, so far as is reasonably practicable, below the DBF thresholds.

This minimum requirement for safety systems is expressed in terms of systems which will be ‘permanently available’ under any permissible plant state. Where it is anticipated that plant operation will be permitted with any systems unavailable (e.g. under maintenance or otherwise out of service), then additional systems must be provided by the design such that at least two will always be available, ensuring adequate protection against DBF sequences in the event of an unrevealed failure.

In providing protection against reasonably foreseeable faults, there are generally accepted hierarchies regarding the way in which hazards are managed and the type of safety systems employed.

The hierarchy which will be adopted for managing hazards is:

(a) Eliminate the hazard;

(b) Reduce the hazard;

(c) Isolate persons from the hazard;

(d) Control the hazard by use of suitable arrangements;

(e) Protect against the hazard by providing mitigating systems.

For hazards which cannot be eliminated, the following hierarchy will be used in selecting the preferred safety measures to control and protect against faults that realise the potential hazards:

(a) Passive safety measures (e.g. package or cell containment; permanent shield walls);

(b) Automatically initiated active engineered safety measures (i.e. no operator action required);

(c) Operator initiated active engineered safety measures (i.e. brought into service manually);

(d) Administrative measures (e.g. manual access controls; evacuation procedures);

As far as is possible, ‘preventative’ systems, which will prevent faults progressing to the point where exposure to elevated radiation levels or a release of radioactive material occurs, will be selected in preference to ‘mitigating’ systems, which reduce the level of worker or public exposure but act only after the event has occurred.

Since it is not always possible to completely eliminate a fault, requirements have been specified for individual safety systems in order to determine whether the control systems and protective measures in place to protect against an individual fault sequence will be sufficient to reduce doses to acceptable levels.

The RPPM [26] requires that the safety measures on any new nuclear facility should, as a minimum, be capable of reducing the consequences of any DBF to below the BSL and, preferably to below the BSO. The fact that the BSO is set at a very low dose level re-emphasises the preference for preventative safety measures (i.e. measures which stop a fault sequence before any significant dose is delivered) over mitigating safety measures.

Section 5.4.4 discusses the safety measures considered to date for the DBFs identified and the possible ways in which the design could be developed to eliminate these faults or mitigate the radiological consequences should they occur. In this case it is the protected doses which are compared against the BSO and BSL targets – that is, the doses which would result if all safety systems identified in the assessment to date, function as intended.

### 5.4.3 Consequence assessment

For the consequence assessment for the deterministic case (i.e. for DBA analysis), conservative assumptions are used, including that the ‘worst-case’ inventories are involved in the fault. For these purposes, a review was undertaken of the waste streams for which processing and packaging arrangements have been proposed by the waste producers under the Letter of Compliance process. The review considered the radioactive inventories in the packages and the robustness of the waste packages against fire and impact challenges. On that basis, ten representative ILW/LLW waste streams were selected as comprising a reasonable set of higher risk waste packages suitable for our current fault assessment work. The ten cases selected cover a range of waste streams and both shielded and unshielded package types. At a later stage we will do further work to identify definitive bounding waste streams.

Eight waste streams involve unshielded packages and two relate to shielded packages. For each fault involving an unshielded package, the worst result from the 8 waste streams has been used in the subsequent assessment. Similarly for the shielded packages, the worst of the two waste stream results has been used.

Three representative HLW/SF/Pu waste streams have also been considered, namely: vitrified high level waste; PWR spent fuel and Pu stocks from AGR fuel reprocessing.

The radiological consequences resulting from individual faults have been estimated for these waste streams using the Repository Operational Safety Assessment (ROSA) Toolkit – a software tool developed by RWMD to assess the radiological doses to workers and the public from accidents. The Toolkit estimates external exposure from direct radiation arising from shielding faults (including loss of shielding following impact events) and internal exposure due to the release of airborne activity during impact and/or fire events involving waste packages. Faults involving resuspension of surface contamination are also considered. However, we note the assessment of this last group of faults indicated that their consequences were very low and therefore they could not contribute significantly to the facility risk. They were therefore excluded from detailed risk assessment.

For this assessment, both partially protected and protected doses have been calculated. As described above, the former takes no account of active control measures such as interlocks on shield door access, ventilation systems or alarms although they do allow for purely passive safety barriers such as cell containments. The partially protected doses have been used to identify potential DBFs. The protected doses are assessed on the basis that all safety measures function as intended. However, it is emphasised that, because this is only an illustrative concept example, no design has yet been provided for many of the safety systems which would normally be present in a modern nuclear installation and hence the results are inevitably conservative. Nevertheless, for those faults identified as DBFs, the protected doses have been analysed to gain an initial insight into the effectiveness of those safety systems identified to date.

The methodology employed in the ROSA Toolkit is described in [10,43]. The Toolkit methodology and implementation has been subject to independent peer review, which included an audit of the verification testing carried out during the development process. We have also carried out acceptance testing on issued versions. Further information is provided in [43].

Full details of the consequence assessment are provided in [8].

### 5.4.4 Discussion of design basis faults

A number of potential faults did not require classification as DBFs because their consequences were too low. This included all contamination faults. Due to their low harm potential these faults do not require further consideration in the DBA analysis.

The most significant DBFs are summarised below:

| **Unshielded packages (UILW)** |
|---|
| Inadvertent exposure of maintenance worker to unshielded package |
| Impacts underground involving a single unshielded package – these faults are only potentially significant in terms of worker exposure, results for the public are fully compliant with dose targets |
| Severe impacts underground involving multiple unshielded packages |
| Fires underground involving unshielded packages |

| Shielded packages (SILW) |
|---|
| Impacts at the surface involving shielded packages |
| Sustained fires (including those following an impact event) at the surface |
| Severe impacts underground involving multiple shielded packages – these faults are only potentially significant in terms of worker exposure, public exposure from these faults is negligible and fully compliant with dose targets |
| Sustained fires (including those following an impact event) occurring underground |
| **HLW/SF packages** |
| Once the protection afforded by the robust packaging of these wastes is taken into account, all faults have been assessed as being fully compliant with all dose targets |

The faults pertaining to each package type are discussed in more detail in the sub-sections below. As described in Section 5.4.3, the following analysis is based on the protected doses estimated for each DBF, since this allows us to distinguish where currently identified safety systems already control risks to an acceptable level and where there remain potentially important issues to be considered during future design development.

### Unshielded ILW packages

*Faults involving unshielded packages in SWTCs*

Up until the point at which UILW packages have been transferred to shielded containment underground, they are protected by the shielded Type B SWTCs, in which they are delivered to site. These are robust containers designed to prevent any significant release of activity or loss of shielding from all but the most severe impact or fire events. Specifically they are designed to withstand drops of up to 9m onto an unyielding surface and fires resulting in temperatures of 800$^{o}$C and lasting 30 minutes.

It is considered that the SWTCs provide sufficient protection to prevent any significant radiological hazard to persons for all credible impact and fire events. This includes all movement and handling of the SWTC on the surface from the point of receipt, its transfer from lorry or rail wagon to a drift wagon and subsequent transfer via a drift to the underground reception area.

Thus, for these operations, the primary means of controlling fire and impact hazards will be the use of the qualified SWTCs and ensuring, by design, that the severity of potential faults will not exceed the Type B qualification criteria. This means, for example, restricting maximum drop heights to less than 9m and minimising fire loadings so far as possible.

Since the SWTC represents a passive safety feature which reduces the partially protected doses below the DBF threshold, faults involving waste packages in SWTCs have not been identified as DBFs. However, in order to highlight the importance of the transport containers to the deterministic safety case, the set of faults involving impact or thermal challenges to SWTCs is listed in [8].

It is further noted that all lifting devices and their supporting structures would be qualified against the design basis events for internal and external hazards. In addition, they would employ fail-safe load protection, such that, if these events were to qualify as DBFs, the design basis criteria would be met.

While the assessment indicates that the qualification of transport packages would, on its own, be sufficient to prevent any significant exposures due to containment or shielding damage under fault conditions, there is the additional requirement to maintain risks (radiological and conventional) as low as reasonably practicable (ALARP). During future development stages, the design will be examined to determine whether there are other options which would help to restrict the frequency of fire and impact events and thereby also reduce the potential for non-radiological damage and injury.

For example, potential means of reducing the frequency of impact faults could include designing the package transfer system in the surface package transfer facilities and in the underground reception area to avoid lifts wherever possible and minimising the height of any lift which is unavoidable. Where it is considered necessary to move packages by crane, the crane design would include safety mechanisms such as emergency brakes and overload protection.

The only significant external radiation fault for which demonstration of compliance with the 20 mSv BSL cannot yet be demonstrated is the inadvertent exposure of a maintenance worker to a bare UILW package in a SWTC which was believed to be empty. This could conceivably occur where, for example, a SWTC had been returned to the surface without having had the waste package removed.

Currently, the only engineered safety system allowed for in this example assessment is installed gamma monitors at the maintenance facilities. These would be backed up by managerial controls such as radiation surveys on SWTCs received at the maintenance facility. Consideration would also be given to weighing SWTCs at this time. There would also be inventory accountancy controls for tracking of all waste packages which would be designed to prevent a package being mistakenly returned to the surface. Another option is to provide an additional engineered safety system, such as an interlock (possibly based on container weight) to prevent inadvertent export of a loaded SWTC from the inlet cell. This, interlock system alone, could be sufficient to provide full compliance with the 20 mSv BSL and, as an engineered safety system not requiring operator initiation, it ranks highly on the hierarchy of preferred safety measures.

*Faults involving unshielded packages underground*

For some of the impact faults involving unshielded packages in the underground facilities, it is not currently possible to demonstrate compliance with the 20 mSv BSL for workers (see Section 5.4.2. Potential protected doses to members of the public are, for the most part, insignificant due to the provision of multiple stages of high-efficiency filters in the air extract system.

Lifting devices in nuclear facilities are, as standard, equipped with multiple safety systems to prevent the occurrence of dropped loads. Such systems include degrees of redundancy and diversity appropriate to their safety importance. This means that there must be sufficient redundancy to provide the required level of protection in the event of one system failing, and that more than one type of system must be provided to ensure continued protection in the event of an upstream fault leading to a common cause failure (simultaneous failure of redundant systems of the same type). Full consideration will be given to the number and type of safety systems required when specifying lifting devices for the site-specific design. It is considered that when such design details are available and the impact on potential fault frequency can be taken into account, full compliance with DBA criteria will then be readily achievable (See Table 5).

Another factor here is that we believe the values used to represent that part of the waste package inventory of radioactive material (the release fraction) released in an impact event are cautious [44]. The impact release fractions (RFs) are derived from a mixture of experimental tests, supported by 3-D modelling [44]. The release fractions used in the analyses are based on data pertaining to a 25 m drop and the fraction of activity released is based on the fraction of break-up of a grouted ILW wasteform. It is conservatively assumed that all particulate generated is released and that the activity is distributed homogeneously throughout the package. No allowance is made for any residual containment provided by the stainless steel drum or box or the non-active capping grout.

The RFs for individual faults are based on the 25m drop height data scaled linearly [44] for drop heights of 3m, 5m and 15m as appropriate.

Further, the RF is based on the quantity of particles of size 100 μm or less produced by the impact. However, particulate greater than 10-20 μm is not usually considered to be respirable and therefore does not present an inhalation hazard [45] but can represent a direct radiation hazard.

We have commissioned further research in this area [44] which it is believed will allow more realistic RFs to be determined and hence permit the calculation of more realistic doses. This work will also consider the effects of ageing on RFs.

Other than dropped loads, the assessed impact faults involve structural failures which may be initiated by external events (e.g. earthquakes). In compliance with regulatory expectation, safety-related structures will be qualified against the challenges associated with external hazards to a degree proportionate to the radiological consequences of their failure. This principle may be extended to cover stacks of stored waste packages if future safety analysis shows this to be necessary. Specifically, it is assumed that, where assessed consequences would indicate that a fault should be included within the design basis, structures will be qualified against natural external hazards (such as seismic disturbances with a return period of 10,000 years). The design basis assessment therefore assumes that there will be no significant radiological consequences for events of that severity.

Where threats to structures from internal hazards are identified by the hazard identification process, the structures themselves and/or systems which could be affected by their failure will be qualified to a degree commensurate with the consequences of their failure.

The most significant impact fault is that involving the collapse of a single stack of packages. We have conservatively applied the same RF (for a 15m drop height) to all packages in the stack. This is pessimistic since most of the toppled packages will fall less than 15m.

The example assessment estimates the potential dose to workers in the crane maintenance area which, although it has separate shielding and containment, adjoins the disposal vaults for unshielded packages. This has been done in order to obtain indicative information on the worst possible consequences. Again the assessment is conservative, since it does not allow for the mitigating effect of the ventilation system in drawing contaminated air away from the crane maintenance area or for early evacuation of the workers in response to activity-in-air alarms which are likely to be installed.

Such protective measures will be considered in detail for future assessments. In the intervening period, as described above, further research will be carried out into the fraction of radioactive inventory that could be released following impact damage to a package in order to reduce any unrealistic conservatisms in the assumptions where this is clearly justified. The aim is to ensure that the importance of any particular hazard is not over-emphasised in comparison to others.

The consequences of underground fires involving one or more waste packages have been assessed as exceeding our BSLs for both public and workers (see Table 5) in some cases. It is recognised that there is a large amount of conservatism in this initial assessment. The fraction of radioactivity released from a waste package in a fire (the fire release fraction or RF) is based on experimental and analysis work to determine potential releases of radioactivity from packages exposed to temperatures in the region of 1000°C for an hour [44].

However, the amount of combustible material available to sustain a fire in underground facilities will be minimal. This is especially true for the unshielded package disposal vaults where fire loadings are likely to be limited to small amounts of lubricating oils and hydraulic fluids in the lifting devices and other equipment. These materials will not be sufficient to sustain a fire approaching the intensity and duration of the experiments on which the present fire RFs are based. Indeed, it is considered that the maximum fire duration is more likely to be of the order of a few minutes. Research work in this area is ongoing and a number of areas have been identified where RF values could be refined and where further work could be undertaken to provide a better understanding of the behaviour and performance of waste packages in faults [44].

Another conservatism of particular relevance to public consequences is the current assumption that the entire inventory of any radionuclides which could exist, or be released, in gaseous form, would be released in the event of a fire. In some cases this is an excessively conservative assumption, since the species will be chemically bound to non-volatile substrates. Further, no claim can be made for capture by HEPA filters for gaseous species. However, those radionuclides released could be chemically bound in solid particles, so it is pessimistic to assume that they would be released to atmosphere without any hold-up on the filters in the ventilation system and discharge stack. The relatively high public doses obtained for these faults all relate to a single waste stream which includes a large inventory of chlorine-36, all of which is assumed to be released in a fire. It is intended to give further consideration to the assumptions concerning the release fractions for gaseous radionuclides in future work [44]. We are confident that this work will lead to significantly lower estimated doses.

Nevertheless, this assessment has highlighted the potential importance of restricting the quantity of combustibles in areas where unshielded packages are present and these findings will be used to inform future design development.

### Shielded ILW packages

*Surface impact and fire faults involving shielded packages*

These waste packages would be Type IP2 transport packages which are not required to have high resilience to impact and fire damage due to their strictly limited permissible radioactive inventory. A number of fault scenarios occurring at the surface have been pessimistically identified as DBFs in our example assessment despite having limited inventory relative to the majority of unshielded packages.

A number of these faults are assessed to result in protected doses which exceed the lowest BSL (1 mSv) for off-site exposures. These scenarios involve vehicular impacts (road or rail) with shielded packages in transit onto or across the site. So far as is reasonably practicable, the developed design for the surface layout will seek to remove the impact potential by removing the possibility of a vehicle carrying a package and any other vehicle being present at the same location at the same time. This is likely to require segregation of road and rail routes, the avoidance of cross-over points between routes and possibly interlock systems. The frequency and severity of impacts will be restricted by the fitting of mechanical speed limiters to on-site locomotives and potentially to on-site road tractors. In this way it should be practicable to limit potential impact energies to a level at which any loss of package containment is extremely unlikely.

The example assessment assumes that two potentially severe impact events involving impact damage to an IP2 at the Waste Package Transfer Facility will be prevented by designing out the potential to lift transport packages over other packages and by appropriate qualification of all crane structures against internal and external hazards. Engineering systems such as physical stops on the crane travel range and passive lift height limitation may be provided. The possibility of lifting a package which is still attached to the delivery wagon leading to damage and possible collapse of the crane can be prevented by the provision of overload protection. The frequency of crane failures resulting in dropped load impacts with the facility floor will be reduced by the inclusion of safety mechanisms for arresting package descent prior to impact. Such systems are standard on cranes required to lift 'nuclear loads'.

It is again emphasised that the current assessment employs very conservative impact release fractions for shielded packages and that these, even when adjusted for varying impact severity, are believed to result in an overly conservative assessment of consequences for the group of faults discussed above. In future, the application of more realistic release fractions can be expected to remove some of these faults from the design basis and the low frequency of dropped loads produced by employing high integrity cranes should ensure that any faults which do remain within the design basis will be classified in the lowest frequency bands which have less restrictive DBA limits (see Table 5).

Using the current fire RFs (see above), it is not possible to demonstrate compliance with the 1 mSv BSL for the public for surface fire events. The most likely initiating event for a surface fire is vehicle impact that results in a fire on a road or rail vehicle operating at the surface. Options which will be considered when designing on-site road and rail systems include:

- Physical separation of road and rail traffic;
- Restricting use of any road/rail sector to a single conveyance at any one time;
- Limitation of vehicle fuel tank capacities.

The first two of these options would ensure that the frequency of impact events leading to fire would be suitably restricted whilst the third would effectively remove the possibility of any fire which did occur approaching the duration and severity of the one hour 1000C fire we assume in our analysis. This would reduce the likelihood of a release occurring and limit the magnitude of any release which did occur.

It should be borne in mind that the fire RFs used represent a cautious estimate of the total activity released from a one hour high temperature fire. With a layer of shielding material (e.g. concrete) between the package containers and the wasteform, it will take a significant period of time for the wasteform to experience a significant increase in temperature even for such a fire. There would, of course, be on-site emergency services for rapid response to fire events and so the assumption of a one-hour fire is very pessimistic. This is discussed further in [44].

*Direct radiation faults involving shielded packages*

The shielding in the shielded packages is designed to restrict external radiation levels (and hence doses to workers) to acceptable levels commensurate with normal operating conditions. If there is insufficient shielding then workers would be exposed to higher than expected radiation levels. For DBA assessment a worst-case scenario has been assumed, where an IP2 designed to be lined with 300mm of shielding has no internal shielding. (This is an extreme scenario. A more credible fault is that shielding of insufficient thickness would be provided. In this case, there would still be significant attenuation of the radiation dose rates.)

Nevertheless, the results demonstrate that even for the bounding scenario, the consequences are acceptable, with no direct radiation faults requiring classification as DBFs.

For packages being transported by road and rail, it is barely credible that a significant shielding deficiency would remain undetected until the package had reached the GDF due to the number of checks all packages undergo prior to and during dispatch. We, as operators of the GDF, would also retain an element of control through audit and inspection of consignors' arrangements [29]. In addition, every package will be subject to a radiation survey on receipt. The development of the design of the receipt facilities will involve consideration of the practicability of providing automatic radiation monitoring of packages at site access points.

*Underground impact and fire faults involving shielded packages*

A number of DBFs involving a fire underground are currently assessed to give rise to off-site doses which exceed the 1 mSv BSL but are below the higher BSLs set for less frequent faults (see Table 5). The assessment of the protected doses allows for the multiple filters on the extract ventilation system, which will prevent airborne radioactive particles being released from the discharge stack. However, this safety measure is not effective against gaseous species. It is particularly difficult to estimate the quantities of such radionuclides (or their precursors) which may be present in the waste at the time of packaging as gaseous and volatile material may have been lost before or during the packaging process. We make the conservative assumption that all potentially gaseous active material will be released in gaseous form in a fire. However, much of this material is likely to be chemically bound in non-volatile form. Under these circumstances, the fraction of material released would be much lower and would, to a large degree, be trapped on the filters in the ventilation system. (For example, the waste stream giving rise to doses above the lowest BSL contains graphite containing a significant inventory of carbon-14, all of which is assumed to be released. In practice, graphite is stable at very high temperatures and has already been exposed to high temperatures in a reactor environment and so the release rate is expected to be very low). It is likely, therefore, that the significance of these faults is being significantly over-estimated. Further work being planned to reduce the uncertainties in this area is described in [44].

Nonetheless, the aim of future design development will be to eliminate the potential for any fire of sufficient duration that significant releases from industrial packages could occur. This would be achieved, for example, by strictly limiting the fire loadings in all package handling areas. The illustrative concept example currently assumes the use of a diesel-powered stacker truck for emplacement operations. This may unnecessarily introduce a significant source of fuel into the disposal area for shielded packages. Future options to be explored include the use of an electrically powered truck and the feasibility of emplacement by crane.

A number of impact events have also been identified as DBFs in respect of worker doses. These events involved either a package drop from the maximum height during stacking or impact damage to several packages (e.g. due to a stack topple). In some cases, these events give rise to worker doses above the 20 mSv BSL [8].

However, as discussed in the subsection on surface impacts, it is believed that the apparent non-compliance with the most restrictive BSLs is a consequence of the conservative release fraction data currently employed. The planned development work in this area is expected to produce a significant reduction in assessed consequences for impacts on shielded packages, removing most such faults from the design basis.

In developing the site-specific design, consideration will be given to potential means of eliminating or reducing the harm potential of these faults. Use of safeguards such as warning sensors on the stacker truck would help to reduce the frequency of impact faults. Alternatively, those faults initiated by stacker truck, due either to human error or mechanical failure, could be eliminated by emplacing packages by crane. This would introduce new potential impact faults but these would be expected to be of lower frequency, given the inherent safety features which could be designed into the crane to prevent non-recoverable human errors. Similarly a stack support system could be a means of preventing stack collapse. Overall, it is anticipated that the provision of preventative controls or redesign of the means of package transfer, will significantly reduce the frequency of underground impacts on shielded packages such that, even if some events remain within the design basis, they will fall into the lowest frequency bands. Our analysis has already shown that the consequences from these events, although above the highest frequency dose target (1 mSv), can be reduced to below the DBA target doses for the lower frequency bands [8]. Consequently, the provision of additional safeguards, as indicated above, would reduce the frequency of these faults and this would be expected to bring their dose consequences below the DBA target doses.

### HLW/SF/PU/HEU packages

*Surface impact and fire faults*

In the illustrative concept example for higher strength rock, it is assumed that all HLW/SF/PU/HEU packages will be delivered to a GDF in copper disposal canisters in transport containers of sufficient robustness that the combined package configuration comprises an approved Type B package. As with unshielded packages, the fact that the wastes remain in their transport package until transferred to shielded underground facilities is a key passive safety measure. Without such protection, serious mechanical impact or thermal challenges to the disposal canisters could potentially have severe consequences. When the robustness of the combined package configuration, is taken into account however, these faults do not result in any loss of containment or shielding and hence do not constitute DBFs.

*Direct radiation faults*

The unprotected direct radiation dose from disposal canister faults was found to be below the DBA threshold criteria; hence no external exposure faults were identified as DBFs. This is due to the very significant gamma shielding afforded by the cast iron and copper components of the disposal canister.

*Underground fire and impact faults*

Following transfer underground, the HLW/SF/PU/HEU disposal canisters are removed from their DCTC for disposal in the deposition holes. In this configuration, they are arguably more vulnerable to fire and impact faults. However, fire and impact assessments have shown that the robustness of the copper disposal canisters will be sufficient to prevent any containment or shielding damage in the event of either a high intensity fire (1000$^{o}$C for 1 hour) or a drop from 8m or less. Therefore, if drop heights are restricted by design to less than 8m (allowing a suitable margin for safety) and other potential impacts are restricted to lesser challenges than such a drop would represent, there would be no radiological consequences arising from mechanical impact challenges to the HLW/SF disposal canisters.

Similarly, given that, for the illustrative concept example, there are no locations in the facility complex which could conceivably support a fire of greater severity than that considered in the assessment, it may be assumed that no release of activity will result from any of the identified thermal challenge faults.

## 5.5 Probabilistic assessment

### 5.5.1 Probabilistic safety assessment criteria

Ideally probabilistic safety assessment (PSA) should be based on data specific to the intended design and hence requires an input of engineering and operational knowledge and judgement. The current concept examples are not sufficiently well-defined to permit risks to be assessed definitively. Nevertheless, it is believed that sufficient information is available in relation to GDF facilities and the way in which they would be operated, to permit preliminary estimates of risk to be made and to commence the development of a PSA methodology.

The Radiological Protection Policy Manual includes criteria against which the output of the PSA process may be judged [26]. These are consistent with regulatory expectation and UK nuclear industry and international accepted practice.

Risk-based criteria are provided for fault conditions which may affect either workers or members of the public or both. For the public, criteria are provided in terms of risk to individuals and societal risk. In each case the criteria are defined in terms of BSLs and

BSOs as defined in Section 3. Below the BSL, risks are tolerable only if they have been reduced to levels which are ALARP. Thus, as discussed in Section 3, the BSL represents the threshold at which the risks from faults become unacceptable. At the very least, risks from new facilities are expected to be restricted below that level. However, in meeting the BSLs, risks may not necessarily be ALARP. The ALARP principle will be applied throughout the development of the design to identify reasonably practicable means of reducing risks still further. The BSO represents the benchmark reflecting modern nuclear industry standards in relation to what should be achievable in terms of restricting risk.

The RPPM criteria relating to the tolerability of risk from faults are presented in Table 6 below.

### Table 6 Risk/frequency targets for PSA

| | | |
|---|---|---|
| **Members of the public** | | |
| Total risk of death to a person outside the site from accidents on the facilities on the whole site which lead to exposure to ionising radiation | **Risk ($y^{-1}$)** | |
| | **BSL** | **BSO** |
| | $10^{-4}$ | $10^{-6}$ |
| Total predicted frequencies of accidents on the plant which could give rise to doses in specified dose range | | |
| **Dose Range (mSv)** | **Frequency ($y^{-1}$)** | |
| | **BSL** | **BSO** |
| 0.1 - 1 | 1 | $10^{-2}$ |
| 1 - 10 | $10^{-1}$ | $10^{-3}$ |
| 10 - 100 | $10^{-2}$ | $10^{-4}$ |
| 100 - 1000 | $10^{-3}$ | $10^{-5}$ |
| > 1000 | $10^{-4}$ | $10^{-6}$ |
| **Workers** | | |
| Total individual predicted risk of death (early and delayed) to a person on the site from on-site accidents that result in exposure to ionising radiation | **Risk ($y^{-1}$)** | |
| | **BSL** | **BSO** |
| | $10^{-4}$ | $10^{-6}$ |
| Frequencies of any **single accident on the plant** which could give rise to doses in specified dose range | | |
| **Dose Range (mSv)** | **Frequency ($y^{-1}$)** | |
| | **BSL** | **BSO** |
| 2- 20 | $10^{-1}$ | $10^{-3}$ |
| 20 - 200 | $10^{-2}$ | $10^{-4}$ |
| 200 - 2000 | $10^{-3}$ | $10^{-5}$ |
| > 2000 | $10^{-4}$ | $10^{-6}$ |
| **Societal risk** | | |
| Total risk of 100 or more fatalities, either immediate or eventual, from on-site accidents that result in exposure to ionising radiation | **Frequency ($y^{-1}$)** | |
| | **BSL** | **BSO** |
| | $10^{-5}$ | $10^{-7}$ |

### 5.5.2 Consequence assessment

PSA is intended to supplement and support the deterministic analysis. Among other things, it provides the means to derive numerical estimates of risk for comparison to the targets discussed in Section 5.5.1 and demonstrates that no single class of fault makes a disproportionate contribution to risk. To enable meaningful comparisons, PSA should be based on best-estimate data and assumptions so far as possible (rather than the worst-case, conservative approach employed in DBA).

For PSA, representative waste streams have been identified for each waste class (i.e. UILW/DNLEU, SILW/LLW or HLW/SF/Pu), based on the radiotoxicity of the package inventory and the number of packages in the waste stream. This approach was used to identify the waste streams in each waste class which are likely to dominate the facility risk.

The waste streams selected for this preliminary analysis are:

- encapsulated fuel cladding in unshielded $3m^3$ boxes;
- graphite decommissioning waste in shielded 4m boxes;
- DNLEU;
- vitrified HLW;
- AGR spent fuel;
- Pu stocks from AGR fuel reprocessing;
- encapsulated HEU.

The ROSA Toolkit, used for the DBA consequence assessment, also permits the calculation of best estimate consequences to workers and members of the public. This is achieved by using average radionuclide inventory and dose rate data as input to the Toolkit.

### 5.5.3 Frequency analysis

For each fault listed in the fault schedule, the initiating event is that which challenges the safety systems in place to prevent or mitigate radiological consequences. Depending on whether or not the safety systems deliver their safety function, the fault may develop by different routes (fault sequences) to a safe or unsafe outcome. Generally, therefore, the assessment of fault sequence frequencies requires the combination of initiating event frequencies with safety system failure probabilities. Since detailed designs of safety systems and equipment employed in operational safety systems are not available, generic data, based on historical data for similar items on other UK nuclear sites, have been used for the assessment of engineering failure probabilities. Where human failures contribute to the fault sequence frequency, these have been derived on a conservative basis, employing recognised human reliability assessment techniques or databases. Further details of the frequency assessment carried out for PSA are provided in [8].

### 5.5.4 Risk assessment

The risk from a single fault is a combination of the frequency (likelihood) of the fault occurring, the radiological dose incurred by an individual as a result of the fault and a risk factor (the probability of death from radiological health effects per unit dose).

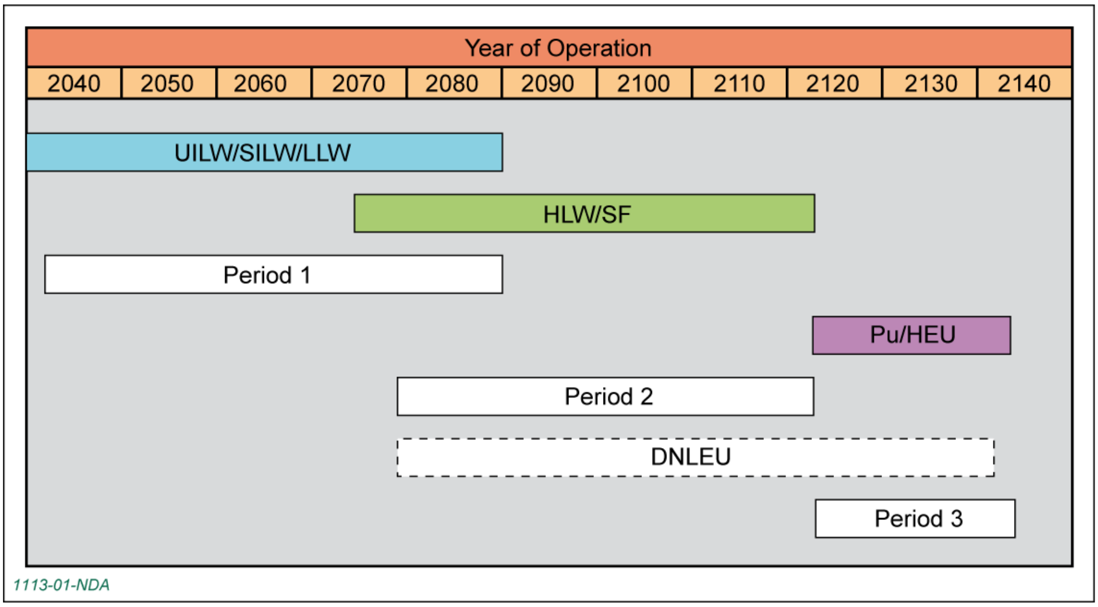
The total risk from the facility is then simply the sum of the risks from all faults on the facility.

Appendix B presents the results of the risk assessment and compares results against the risk targets. Three separate time periods have been considered based on a provisional waste emplacement schedule. Period 1 covers 2040 – 2090, from the commencement of waste handling operations to the completion of ILW/LLW package emplacement. It also includes some HLW/SF emplacement operations, since these are currently assumed to commence in 2075. Period 2 comprises 2080 – 2120, when HLW/SF emplacement operations are assumed to be completed. Finally Period 3, 2120 – 2140, covers all Pu/HEU emplacement operations. It has been assumed that DNLEU packages will be delivered in parallel with the HLW/SF and Pu/HEU packages.

This timeline is depicted schematically in Figure 11.

The most significant assessed risks in each of these Periods are summarised below.

**Figure 11 Assumed timeline for receipt of wastes**




### Risks for period 1

Period 1 risk is taken to be the sum of risks incurring as a result of ILW/LLW and HLW/SF package emplacement and storage, since these waste types may be delivered in parallel over this period. The results are summarised in Table B1 in Appendix B.

*Public risk*

The total risk to the most exposed individual member of the public from all faults on the facility is assessed to be only a very small fraction of the BSO of $10^{-6}$ per year (see Section 3). The risk is dominated by faults involving unshielded packages. (Note that the “most exposed individual member of the public” is normally defined as someone living in the nearest human habitation or at a distance of 1kilometre from the facility if that is nearer to the site. Since we do not have a specific site, we have assumed for this analysis that the most exposed member of the public is located at 0.7 kilometres from the GDF.)

It may also be seen from Table B1 (Appendix B) that the public risk is dominated by faults of relatively low consequence and particularly by those with consequences below 1 mSv.

*Worker risk*

For workers, the total risk from all faults on the facility equals the BSO of $10^{-6}$ per year. However, for conservatism, this assessment adds the risks from each fault as if the same individual would be exposed by each fault. In practice, this is unlikely to be the case. For example, in some cases, transport personnel would be exposed while in others maintenance operators would be exposed. Clearly, the risk to any single employee would be very much lower than the assessed figure.

Comparison with the dose-frequency criteria from the RPPM shows that for the 2 to 20 mSv and the 20 to 200 mSv dose bands, the highest estimated fault frequency lies above the BSO but still below the BSL (see Table B1, Appendix B). UILW and SILW wastes contribute roughly equally to the total risk to workers from faults, while the risks from HLW/SF wastes are negligible in comparison. Fire scenarios involving SILW packages, external radiation from UILW packages and impact scenarios involving UILW packages all contribute significantly to the total risk.

### Risks for period 2

For Period 2, it has been assumed that DNLEU packages will be delivered in parallel with the HLW/SF packages (see Figure 11). The total risk for this period is therefore simply the sum of the risks for these waste categories. The results are presented in Table B2 in Appendix B.

*Public risk*

The total risk to a member of the public from all faults during this period has been assessed to be very low, i.e. several orders of magnitude below the BSO. Indeed the risk figures generated by the assessment are well below the level at which risks and frequencies have any real meaning.

The summed estimated frequency of faults in any dose band is also only a very small fraction of the relevant BSO frequency. There are no faults assessed to have doses above 10 mSv. Rather, the public risk is dominated by faults of relatively low consequence and particularly by those with consequences below 0.1 mSv.

*Worker risk*

The total risk to workers from all faults is less than $10^{-8}$ per year, two orders of magnitude below the BSO. The summed frequencies in the individual dose bands (see the worker dose/frequency ladder in Table B2) are a small fraction of the BSO in each case.

Over 70% of the risk to workers from faults comes from DNLEU wastes. Risks from impact events contribute most to the total while external radiation and impact followed by fire scenarios make up the remainder. The impact fault contribution is derived from faults involving DNLEU while HLW/SF wastes provide the bulk of the risk from external radiation faults.

Overall, the worker risk in this period is dominated by faults with consequences below 2 mSv. There are no faults with consequences above 200 mSv.

### Risks for period 3

For Period 3 it is assumed that DNLEU and Pu/HEU packages will be delivered in parallel. Thus the total risk is simply the sum of the risks for these waste categories. The results for Period 3 are given in Table B3 in Appendix B.

*Public risk*

The total risk to the public from all faults during this period is minimal at many orders of magnitude below the BSO.

*Worker risk*

For workers, the total risk from all faults on the facility is again less than $10^{-8}$ per year. The dose/frequency ladder is similar to that for Period 2, with all BSOs being met.

## 5.6 Severe accident assessment

### 5.6.1 Introduction

In this section, we describe our approach to Severe Accident Analysis (SAA) for a GDF.

The DBA analysis (see Section 5.4) examines those faults which are considered to be reasonably foreseeable. In line with regulatory expectations and industry good practice, such faults have been defined as having an initiating event frequency of more than $10^{-5}$ per year (that is – potentially occurring at least once in 100,000 years). Inevitably, possible faults with potentially very serious radiological consequences can be conceived whose likelihood of occurrence is so low as to lie below this threshold. Designing facilities specifically to address such faults is generally not reasonably practicable. For these faults, known as severe accidents, the safety measures introduced as a result of DBA may not be effective. However, the robust application of DBA should ensure that their frequencies are very low and that, overall, the faults should make a relatively small contribution to the facility risk.

Nevertheless, severe accidents still need to be addressed in the safety case. Severe Accident Analysis (SAA) is expected to be performed on a best-estimate basis and to provide estimates of risk as an input to the PSA. It is performed primarily to provide an understanding of the required emergency response to such accidents but should also identify where the provision of additional safety measures and accident management measures and resources may be reasonably practicable.

For designation as severe accidents, events must, as described above,

- have initiating event frequencies below the threshold for inclusion in the design basis, i.e. below $10^{-5}$ per year, and
- be capable of delivering consequences above the BSLs defined for Design Basis Faults for low frequency events, i.e. above 100 mSv off-site and above 500 mSv on-site.

### 5.6.2 Identification of severe accidents

To ascertain whether any faults on a GDF would require to be the subject of SAA, the output of the 'best-estimate' ROSA Toolkit consequence calculations carried out for the waste streams selected for DBA and PSA was reviewed. No fault in any of the waste streams considered for risk assessment gave doses in excess of the SAA thresholds for either workers or the public. These doses were assessed taking account of only those mitigating barriers which would be likely to survive the fault. Thus, for an aircraft crash on the site, no mitigation is assumed for packages on the surface apart from some residual protection attributed to a damaged SWTC. Packages underground, on the other hand, are assumed to be unaffected. Since these waste streams were selected to be the most 'risk dominant' within their waste class, these results tend to support the view that it is unlikely that potential severe accidents will be identified in the course of the development of the safety case for a GDF.

Nevertheless, because of the potential importance of severe accidents within the developing safety case, an open mind will be kept on the possibility of their occurrence throughout the safety case development process.

The highest assessed doses to workers and members of the public were for aircraft crashes onto surface facilities. These doses were assessed by the Toolkit which makes conservative assumptions in regard to how many transport packages will be directly impacted and involved in an aviation fuel fire. In making these assumptions consideration was given to the possible disposition of loaded transport containers relative to the possible aircraft impact directions and the location of incompressible aircraft features such as jet engine shafts. The assessed on-site doses were highest for the HLW and Pu waste categories while the highest off-site doses assessed were for unshielded ILW packages in SWTCs. Even on the basis of pessimistic assumptions in regard to the residual containment afforded by Type B transport packages in an aircraft impact, the highest assessed worker and public doses calculated were 285 mSv and 7 mSv respectively.

These faults have not been identified as severe accidents at this time. However, given the considerable uncertainty associated with the estimation of consequences of aircraft impact accidents, the following paragraphs consider the potential implications of such accidents for design of a GDF and emergency response arrangements.

### 5.6.3 Potential design implications

Accidental aircraft impact represents the most severe accidents which could occur at the surface facilities of a GDF. (Note that deliberate aircraft impacts are a security issue and are not discussed or analysed in the generic OSC.)They are characterised by the potential for direct impacts on transport packages which may well be beyond the design basis of even Type B packages and certainly beyond that of the IP2 shielded packages. It is also highly likely that such events would involve a very intense fire fuelled by aviation fuel which, again, would be beyond the transport packages' design basis.

It is emphasised that any such accidents are predicted to occur on a frequency of less than once in 10 million years due to the relatively small effective target area presented by groups of packages in temporary storage following their receipt on-site.

Clearly, the frequency of these accidents would normally be too low to justify their being addressed directly in the development of the facility design. However, the requirement to provide any further protective measures which are reasonably practicable to implement remains.

As always, the first consideration would be to prevent the accidents entirely. It is standard UK practice to establish no-fly zones around nuclear facilities. Clearly this would further reduce the already low frequency of such accidents but could not be guaranteed to completely remove the possibility of their occurrence.

To completely remove the possibility of aircraft impacts directly onto transport packages would require that all package handling and temporary storage facilities currently located at the surface horizon be relocated underground. While this option is worthy of consideration, it is highly unlikely to prove to be reasonably practicable since the costs associated with its implementation would almost certainly outweigh the removal of what is, in fact, a very small risk.

### 5.6.4 Implications for emergency response

The fire and rescue service resource required to respond to aircraft impacts on the site would be similar to that required for any aircraft accident in an inhabited area but with the added requirement for specialist resource trained in radiation protection. Any accident involving larger transport or passenger aircraft would require fire fighting resources beyond the level normally provided by way of dedicated on-site fire and rescue personnel and equipment. However, it is extremely unlikely that any increase above this level, located on-site, specifically to deal with aircraft impacts could be justified given the very low likelihood of such accidents and the small risk reduction that such an increase in provision could achieve.

The type of measure which could be implemented, however, is to avoid the potential for an aircraft crash to be a common cause for the initiation of a major on-site radiation accident and the disruption of the fire and rescue capability. This could be achieved by ensuring that the fire and rescue station is located well away from any area in which transport packages would be stored or processed. This is recognised good practice on nuclear sites and has already been taken into account in the example surface layout for a GDF concept. Measures such as these will be taken forward to the site-specific design.

## 5.7 Criticality faults

The generic OSC includes an assessment of the risk of criticality events during the operation of a GDF. A criticality event is an unplanned and uncontrolled fission chain reaction that results in a sudden very large release of energy and radiation. It can only occur when fissile material is present. A criticality event can cause structural damage and result in potentially lethal doses of radiation to those in the immediate vicinity and warrants careful consideration. A separate Criticality Safety Assessment has therefore been undertaken [9].

The analysis has shown that the only potential challenges to the criticality safety during the operational phase lie in failure by waste packagers to control package fissile material inventories to comply with limits. While this failure on the part of the waste packagers has been identified as a potential cause of criticality, this does not mean that we do not recognise our responsibility as a potential licensee to restrict the criticality risk at a GDF to ALARP levels.

In conjunction with waste packagers, RWMD has established safe fissile masses for generic waste package types by provision of generic criticality safety assessments. These are based on pessimistic assumptions regarding waste characteristics, waste package design and emplacement arrangements [28]. In some cases waste packagers may wish to work to higher limits which have to be justified on a case by case basis. The Letter of Compliance disposability assessment process provides the mechanism for checking that the generic cases have been appropriately applied, or for developing the package-specific case in the event that the generic case is either not applicable or should the packager wish to develop a case to work to a higher limit. The LoC assessment process requires waste packagers to develop Criticality Compliance Assurance Documentation, in which they demonstrate how their procedures ensure that the safe fissile masses applicable to those waste packages will not be exceeded. An important aspect of the arrangements for preventing criticality is auditing of the waste packagers' systems and procedures for the control of the fissile material content of waste packages.

The total estimated risk to the operator from criticality during the operating phase is $1.2 \times 10^{-7}$ per year [9]. The RPPM [26] does not set separate risk criteria for criticality faults and so these risks must be added to the total risk from other types of radiological fault for comparison with the BSL and BSO defined in the RPPM for worker risk. For Period 1 (2040-2090), which covers all ILW/LLW and some HLW/SF emplacement operations, inclusion of criticality faults brings the total worker risk from all faults to $1.1 \times 10^{-6}$ per year which is marginally above the BSO target value of $10^{-6}$ per year. This is still well below the BSL target value of $10^{-4}$ per year. For the remainder of the operational period, the worker risk for all other radiological faults is well below $10^{-7}$ per year. Therefore, the total worker risk, including criticality risk, is below the BSO.

Expressed as an individual risk, the public risk from criticality faults is also significantly below $10^{-7}$ per year. The total annual public risk from all radiological faults is therefore well below the BSO throughout the entire operational period.

The risk to both workers and the public is entirely due to fault sequences involving overbatching. Evaluation of the frequency of overbatching events that are sufficiently severe that there is a possibility of criticality during GDF operations is not straightforward. The Criticality Safety Assessment includes qualitative arguments to support the belief that significant overbatching events are very unlikely. However, further work will be undertaken to examine the procedures and processes at the waste packagers' plants with the aim of quantifying the likelihood of overbatching.

The Criticality Safety Assessment has shown that normal operations cannot give rise to a criticality incident. Calculations have shown that, even under pessimistic assumptions with regard to the distribution of fissile material within packages, expected arrangements of waste packages will be safely sub-critical during all normal operations in a GDF.

The assessment also demonstrates that a Criticality Incident Detection and Alarm System would not be required for the GDF [9].

Overall, the Criticality Safety Assessment concludes that the overall risk due to criticality events during normal operations and under fault conditions is very small [9].

## 5.8 Implications for illustrative concept examples in different geologies

This example assessment has largely been based on the concept example for higher strength rock. However, this does not mean that the chosen site will exhibit this type of geology. For the generic DSSC, we have, to typify the range of potential host rocks in the UK, developed concept examples for two other geologies, namely: lower strength sedimentary rock and evaporite rock [33]. The concepts are described in Section 4.

Apart from the notable exception of the potential use of a shaft for transferring waste packages underground in the evaporite concept example, it is considered that, of the three geologies, the concept example for higher strength rock is bounding in terms of the potential consequences of impact and fire faults. This is due to the larger dimensions of the vault which leads to increased stack heights and associated drop heights. The frequency of dropped loads and impact events could conceivably be argued to be greater for those concepts employing stacker trucks rather than cranes for emplacement, but this cannot be established definitively at this stage.

With regard to external radiation faults, the current concept examples potentially give rise to different exposure geometries (due to differences in underground facility dimensions) and hence different worker doses. The consequences of such faults will be very dependent on the actual characteristics of the selected site, including the host rock. The use of additional shield doors at the entrances to the disposal vaults and tunnels in the concept examples for lower strength sedimentary rock and evaporites will increase the potential for external exposure faults due to failure of access controls, although these will not necessarily lead to greater consequences. In any case, the current assessment indicates that these faults tend to be relatively insignificant in terms of consequences and risks compared to fire and impact faults. Relevant issues will be considered in detail as site-specific designs are developed in line with the site selection process.

Our safety assessment has not identified any safety issues in any of the illustrative concept examples that would either rule out or seriously disadvantage any one concept as compared to the others.

Further comment on operational safety associated with the different concepts is provided in the ALARP discussion in Section 5.10.

## 5.9 Implications of Upper Inventory

The Upper Inventory identifies the potential for increased volumes for all types of waste and materials. Although the Upper Inventory includes new build SF and ILW (assuming no reprocessing of new build SF), no significantly different waste types are introduced in this inventory.

The SF from new build plant is likely to have a higher average burn-up than the SF in the Derived Inventory Reference Case. Consequently, assuming the same cooling period (i.e. the time between removing the fuel from the reactor and its arrival at a GDF) and waste package design, the new build SF radiation levels and heat output would be higher. Radiation shielding at a GDF and in the design of SF packages will take into account the higher radiation levels from new build spent fuel. Similarly emplacement equipment and the spacing of waste packages within a GDF would take the higher heat output into consideration.

Our design basis assessment work uses bounding assumptions for waste packages in the faults examined. These bounding assessments are based on Baseline Inventory waste streams that include SF, where the radionuclide inventories are pessimistically calculated assuming a burn-up that is consistent with that expected for new build. Consequently, if we were to perform a bounding operational safety assessment for the Upper Inventory there would be no new SF faults to consider and the bounding consequences would be broadly similar to those obtained from the Baseline Inventory assessment. The SF fault consequences that could change are those involving exposure of workers to direct radiation due to the anticipated higher external dose rates for the new build SF canisters. However the cooling period is likely to be a more important factor in determining precise consequences than the higher burn-up of the fuel.

At present there is no detailed information on new build ILW/LLW waste so the approach adopted, as outlined in [18], is to assume the bulk material compositions and material grades for the corresponding Sizewell B waste streams and, where these are not specified, to use nominal assignments based on other information in the ILW/LLW derived inventory. This means that any assessment of new build ILW/LLW would not result in higher dose consequences. We would expect that the most limiting ILW/LLW package types would remain the same.

Our probabilistic safety assessment work (see Section 5.5) is based upon “representative” rather than bounding waste streams. These too would not change for the period up to the end of emplacement of fuel from existing reactors, if we assumed the Upper rather than the Baseline Inventory.

However, when new build SF emplacement would be carried out, the ‘representative’ SF waste stream would be likely to change and the adoption of an Upper Inventory could alter the consequences and the risks assessed for SF wastes for a small number of faults (although it would not change the type of fault assessed). The faults that could change are those involving exposure of workers to direct radiation and the aircraft impact fault. However, as for the DBA analysis, the cooling period is likely to be a more important factor in determining consequences than the higher burn-up of the fuel. Any changes to fault consequences are expected to be small.

As can be seen from the analysis presented in Section 5.5, the risks from SF waste streams are likely to be very small in comparison with the contribution from new build unshielded ILW waste streams emplaced in parallel because the massively strong construction of the disposal canisters for SF makes them robust against any credible mechanical or thermal challenge.

Thus our overall conclusion is that, if we were to consider the Upper rather than the Baseline Inventory, the results of our assessment as regards facility risk would be unlikely to be significantly affected.

The Upper Inventory would involve the disposal of substantially more waste and therefore the period of operation of a GDF would be extended making the integrated risk over the GDF lifetime higher. However, regulatory limits and targets are set on an annual basis so an extended period of operation has no effect on the acceptability of our safety assessment conclusions. There are no regulatory limits or targets set for the lifetime risk of a nuclear facility. The NDA specifications for waste packages [29] recommend that ILW packages be designed to maintain their integrity for a target period of 500 years. Hence extending the period of operation of a GDF to allow for disposal of a larger inventory would not effect the analysis appreciably, either in terms of waste package fault performance (and hence consequences), or in terms of the frequency of faults initiated by package structural failure.

## 5.10 ALARP discussion

ALARP studies essentially concern the identification and review of options which might further reduce risks, in order to identify those which would be reasonably practicable to implement through the design process. Definitive ALARP studies cannot be carried out at this early stage, prior to site selection.

Nevertheless, as is its purpose at this early stage in the development process, the DBA analysis presented above has highlighted a number of potential design issues relevant to a GDF concept. When review and revision of the DBA work is complete (see Section 5.4.2), consideration will be given to the potential requirements for additional safety features in the design. Any safety requirements that are identified will be recorded as SFRs (see Section 3.1.3). The types of safety requirements and associated design solutions that may arise as the design and safety case develop are discussed below.

### 5.10.1 Dropped load faults

In line with the design safety principles, the main aims of the design in relation to the prevention and mitigation of dropped load faults involving waste packages will be the:

- Avoidance of unnecessary lifts;
- Minimisation of the height of unavoidable lifts;
- Avoidance of lifts of waste packages or transport packages over other packages;
- Provision of means of raising packages which support packages from beneath rather than requiring their suspension;
- Provision of intrinsic, passive fail-safe safety features on lifting devices.

In the illustrative concept examples, one of the options whose practicability has been considered in relation to the avoidance or minimisation of lifts is the siting of the inlet cell and the disposal vaults for unshielded packages on the same horizon. In a revision to a previous PGRC design [35], the concept example for the higher strength host rock now places the transfer tunnel to the disposal vaults on the same level as the inlet cell. Packages are transferred to the disposal vault through a hatch in the transfer tunnel roof [33]. This removes the lifting operation from the inlet cell to the transfer tunnel and reduces the lifting height for the transfer from the transfer tunnel to the vault. In practice, the design will depend on the characteristics of the actual site chosen, but this demonstrates the features which could be considered in developing the site specific design.

In the illustrative concept examples for the lower strength sedimentary and evaporite host rocks, the Inlet Cell and the UILW vaults have been located on the same horizon eliminating a potentially significant design basis fault.

Other areas where the need for lifts could potentially be eliminated by design are:

- Waste Package Transfer Facility;
- Underground reception area for unshielded packages;
- Underground reception area and buffer store for shielded packages.

One potential option would be to slide packages onto/off rail wagons rather than lifting them by crane. Once unloaded, they could be transferred to their destination on a low level bogie on rails.

Failure of cranes and slinging equipment could lead to the uncontrolled lowering of suspended loads. An alternative approach would be to use hydraulic lifts to support loads as an alternative to cranes. Specifically, for the example case, this approach could potentially be applied to eliminate at least some lifts in the inlet cell and lifts within the HLW/SF disposal area.

Where there is no practicable alternative to lifting by crane or hoist, all reasonably practicable means for preventing or terminating uncontrolled lower fault sequences will be implemented irrespective of the impact qualification of the load. Potential crane safety features which will be considered include:

- Diverse load paths;
- Diverse emergency braking systems;
- Fail-safe manual controls;
- Fail-safe on power loss;
- Mechanical limitation of travel speed;
- Physical limitations on crane travel;
- Capability of safely setting down a suspended load following any fault or external hazard.

Another contribution to safety will be the high integrity of the lifting features on waste packages. The lifting grapples for raising waste packages will also be required to be of high integrity.

One issue specific to cylindrical transport containers, particularly DCTCs, is the possibility that they will require to be rotated within shielded facilities to permit the unloading of disposal canisters. There is potential for toppling or dropping during such operations. It is also noted that vertical unloading (and hence the need for rotation) will increase the height of underground facilities thereby increasing potential drop heights. In a site-specific design for HLW/SF disposal, important issues to consider will be:

- Avoidance of the need for rotation to the vertical through use of horizontal unloading;
- Employment of tilting frames rather than cranes for rotation to the vertical.

The illustrative concept examples for the lower strength sedimentary and evaporite rocks do, in fact, avoid the need for vertical unloading because they employ horizontal deposition.

### 5.10.2 Vehicle impact faults

Consideration will be given in future design development to additional design features which might prove reasonably practicable for preventing or mitigating collisions on the surface. A basic requirement will be to segregate the on-site rail and road systems from one another. This is already provided for in the outline surface layouts for the illustrative example concepts. Segregation of road vehicles carrying waste packages from other traffic would be similarly beneficial.

Another potential solution would be to use passive, engineered features on road and rail networks to limit vehicle speeds. This will help to reduce the likelihood of vehicle collisions and limit impact energies of any collisions that do occur. Mechanical speed limitation of on-site locomotives (including rack and pinion locomotives) is already assumed in the illustrative concept examples. To extend this solution to road vehicles would require the employment of a dedicated tractor unit for on-site movement of road trailers.

### 5.10.3 Fire hazards

One of the principal design issues arising from the hazard control discussion is the need for minimisation of fire loadings in all areas handling radioactive inventory, to reduce the likelihood of fire initiation and to limit the severity and duration of combustion, in the unlikely event that a fire does occur.

In the disposal areas for unshielded ILW/LLW packages, fire potential is very limited. One potential fault condition identified to date is a failure of a lifting or transfer device leading to the ignition of hydraulic fluids or lubricants. However, it is considered that the specification of fire resistant lubricants and hydraulic fluids at the detailed design stage will virtually eliminate the potential for this scenario. It is further intended that, so far as practicable, hydraulic equipment will not be used in the vicinity of waste packages.

In the disposal areas for shielded packages, another potential fire initiator identified in the fault schedule is a collision or toppling event involving the stacker truck, leading to the rupture of its diesel fuel tank and spillage and ignition of diesel fuel. This particular fault could be eliminated by the use of an electrically powered stacker truck- an option which would be preferable for underground working. Of course, this will not completely eliminate the potential for fires associated with the emplacement vehicle. Consideration would require to be given to means of minimising, and preferably eliminating, the potential for ignition and combustion of the vehicle's tyres- a significant contributor to the fire loading on many powered vehicles. A further option that may be considered is to radically alter the design of the shielded waste disposal area to permit emplacement of shielded waste packages by overhead crane. It is likely that this would significantly reduce the fire loading associated with the means of emplacement but there are likely to be other, balancing considerations in determining its practicability.

During future design development, consideration will also be given to the prevention of fires involving the drift locomotive through the design of the locomotive (e.g. to minimise the use of flammable materials in its construction), the use of electrical isolation systems and the prescription of regular maintenance of the rack and pinion system to prevent the build-up of grease on the tracks.

For all facilities and vehicles involve in the processing of waste packages, other options which will be considered include:

- Automatic fire detection and suppression systems;
- Isolation of inventory from electrical systems;
- Avoidance or minimisation of the use of potentially flammable materials in-cell and employment of fire resistant forms of electrical cabling.

Vehicle accidents are currently identified as the main potential cause of fires on the surface. The primary hazard here relates to shielded ILW/LLW packages, since all other waste packages will form part of 'Type B' transport packages which, by definition, have very high fire resistance. As discussed in Section 5.10.2 above, consideration will be given to reducing the potential for road and rail vehicle impacts by segregating road and rail networks and, so far as is practicable, segregating the road network for vehicles carrying waste packages from that for other traffic.

Limiting vehicle speeds will also help to reduce the likelihood of fires by reducing impact energies and therefore the likelihood of fuel tank rupture in the event of an accident. It is anticipated that this could largely be achieved by incorporating passive traffic-calming features in the roads design, although active systems such as speed governors on vehicles could also be used where practicable.

An associated aim of the design will be to control the quantities of hydrocarbon fuel available to sustain fires. This may be achieved by:

- Limitation of fuel tank capacities on all hydrocarbon fuel-powered vehicles;
- Optimisation of the location and capacity of on-site fuel storage tanks;

- Control of site access for fuel tankers (or, possibly, piping fuel onto site from off-site storage).

It has also been noted that, under certain circumstances, prevention or limitation of hydrocarbon pool fires could potentially be addressed via the design of drainage systems for areas processing or storing waste inventory.

### 5.10.4 External radiation faults

During the course of this assessment, a number of design issues have been noted in relation to the means of controlling faults which could realise a significant external radiation hazard. These are detailed below by way of illustration.

**Interlock systems**

The example assessment has indicated that potentially the highest exposures could result from the export from the inlet cell or HLW/SF transfer facility of transport containers believed to be empty but actually containing waste packages. In future development of this part of the design, one option for consideration may be some form of interlock system, potentially based on load cells and linked to the computerised waste accountancy system.

For UILW and HLW/SF, waste packages are only removed from the shielding of their transport containers when within heavily shielded containment facilities. In the illustrative concept example, interlock systems controlling access doors to these facilities would be the primary engineered control against operator exposure to unshielded packages. The work to date has indicated that such systems will be necessary in the following facilities:

- inlet cell;
- transfer tunnel leading from inlet cell to disposal vaults;
- disposal vault crane maintenance area;
- HLW/SF disposal areas.

While the concept examples do not currently furnish details of these systems, a number of issues are identified which will require consideration in the design process for these systems. Due to the potentially high consequences of their failure, interlocks systems are likely to require at least two independent channels of protection to ensure suitably high system integrity. For example, Castell Key type systems could be supported by gamma monitor based systems. It is anticipated that this type of diversity would be required to achieve the necessary level of system reliability. It is also noted that if software based systems were employed, diversity in software development could also be required.

Given that a number of interlock systems would be required throughout the facility, consideration is to be given to the development of an interlock philosophy which could be consistently applied across the plant.

**Gamma monitors**

For any GDF concept, a system of installed gamma monitors will certainly be necessary to meet the basic requirements of radiation protection legislation. The development of such a system would need to consider the following:

- Gamma monitoring detection heads would be located directly in line with and as close as is practicable to all points of access to shielded containments.
- Ensuring that installed area gamma monitors would remain suitably independent of any gamma monitors forming part of an interlock system.

**Drive through monitors**

For all forms of waste delivered to the site, there exists the possibility that gamma and/or neutron dose rates may exceed acceptable levels. To ensure early detection of such hazards and prevent the packages being processed through the facility, it may be necessary to consider providing 'drive through' radiation monitors at critical locations such as:

- Rail and Road Gate Receipt Facilities;
- WPTF access doors.

An alternative or additional measure would be to carry out manual radiation monitoring of transport packages before they were accepted onto the site.

### 5.10.5 Shielded package emplacement

In line with the current illustrative concept examples, the example assessment is based on the use of a stacker truck as the means of transferring shielded packages from a buffer store to their emplacement location within the disposal vaults. A significant number of potential faults have been identified involving this conveyance. These include impacts and fires. Impacts involving shielded packages during package handling operations in the buffer store and disposal vaults also make a significant contribution to worker risk.

The following discussion illustrates the need to consider the balance of advantages and drawbacks associated with alternative design options.

Should there prove to be benefits in retaining the stacker truck as the means of waste package emplacement, potential options for reducing the associated fault frequencies include the incorporation of remote control systems and for the stacker truck to be mounted on a fixed rail system to ensure positive positioning. A remotely-operated stacker truck is considered for emplacement of unshielded packages in the evaporites concept example. For the concept example for lower strength sedimentary rock, the stacker truck only needs to move in a direct line in and out of the disposal vault, so alternative means of positioning, such as warning sensors, may be practicable.

Another possible design option for shielded packages would be to employ a crane for package movement from receipt by rail through buffer storage to emplacement. This approach would have the added advantages of reducing normal operations exposures to radiation from packages, as well as reducing the potential for both fires and impacts. Clearly the use of a crane would introduce a new set of potential faults. However, it is considered that higher reliability could be built into such a system, particularly in terms of removing human error potential.

One disadvantage of the crane emplacement option is that the space in the roof required for the crane would impact on the available space for the waste packages and would be less efficient than the stacker truck in that respect. As vault sizes reduce, the crane takes up a larger proportion of the available space and this would necessitate an increase in the number of vaults for the same volume of waste. For the concept example for lower strength sedimentary rock, where the disposal vaults have shorter operational lives, there is the additional issue of whether to re-use cranes in different disposal modules or provide duplicate equipment. (The evaporites concept example, as currently envisaged, would not utilise crane emplacement).

An alternative risk reduction strategy would be the introduction of a purpose designed transfer bogie on rails onto which the shielded package could be slid from the rail wagon on receipt and transferred to the disposal vault. It may be possible to combine this transfer system with a hydraulic raising system for the final emplacement of packages.

The optimum solution is likely to be highly dependent on the geological characteristics of the selected site.

In developing the safety case, the systematic identification and analysis of potential fault scenarios associated with such alternative design options will play a key role in refining our design to reduce the risk from faults so far as is reasonably practicable.

### 5.10.6 HLW/SF deposition machine

For the example assessment, it is assumed that this machine would be required to be capable of conveying the disposal canister to the deposition hole in a horizontal configuration. It would then be required to be able to rotate the canister to the vertical and lower it into the deposition hole. The canister would require to remain shielded at all times. The potential for crushing the canister during rotation has been identified as a potential fault should the canister be out of position during the rotation phase. However engineered safeguards against such faults are straightforward to design and will be employed where shown to be necessary.

An alternative strategy would be to adopt a horizontal deposition approach in the higher strength rock similar to that currently envisaged for the lower strength sedimentary rock and evaporites concept examples. This would also preclude the possibility of dropping the disposal canister during deposition.

## 5.11 Post-emplacement operations

### 5.11.1 Retrievability

In line with EA guidance [15], the GDF is intended as a disposal facility, rather than a storage facility. This means that it will be designed in such a way that the waste can safely remain in the facility and not require subsequent retrieval. Nevertheless, the DSFS recognises the requirement of the MRWS White paper [1] not to foreclose the option of retrievability and specifies that the design and construction of a GDF should, where possible, be carried out in such a way as not to preclude the option of extended retrievability. This may mean that, for a period of time after the waste emplacement phase, monitoring of the emplaced wastes would be required until the decision is taken to backfill, seal and close. During this time, it would be possible to retrieve the waste in a relatively straightforward manner by reversal of the emplacement operation.

The ease with which retrievability is achievable depends on a range of factors, including the nature of the geological environment in which a GDF is sited. Future decision-making regarding retrievability will need to take account of relevant site-specific characteristics.

In some concepts the waste packages can be emplaced in such a way that they can be retrieved relatively easily at any time until the facility is sealed and closed. This is the case for ILW/LLW waste packages in the illustrative concept example for higher strength rock. The concept currently provides the option for delayed backfilling of the vaults until just prior to closure. This means that the waste packages could be retrieved by a simple reversal of the emplacement process. Assuming no deterioration of package integrity and ability to maintain vaults and handling systems operable, the safeguards in place to protect against fault conditions during emplacement would also protect against faults occurring during retrieval.

Retrieval of disposal canisters from the HLW/SF/Pu/HEU disposal areas in the higher strength rock concept would be less straightforward since the wastes are surrounded by a swelling bentonite clay buffer in the deposition holes and the deposition tunnel above the holes is progressively backfilled as depositions take place.

The same approach to backfilling of the HLW/SF/Pu/HEU disposal areas is adopted in other concept examples, except that the buffer materials used are bentonite pellets and crushed salt for the lower strength sedimentary rock and evaporite concept examples respectively. The buffer material is designed to protect the disposal canisters and prevent or reduce groundwater flow around them. Subsequently, should the canister lose its integrity in the post-closure phase, the buffer also helps to slow down the release of radionuclides.

For the lower strength sedimentary rock and evaporites concept examples, the characteristics of the geologies mean that it is technically more difficult to keep the disposal vaults open for extended periods. Currently, for these concepts, an operational life of 15 years is assumed for each disposal module and associated inlet cell, after which time that module would be backfilled. If the option of retrievability is to be kept open in these rock-types we would need to engineer more permanent structural support than is envisaged in the current conceptual designs.

The potential for cavern failure (due e.g. to rock falls or groundwater ingress) is likely to increase as the timescales over which the facility needs to remain open increase. Backfilling each section immediately following emplacement reduces this risk and provides physical protection for the waste packages. It also avoids the potential risks to workers associated with extensive monitoring and maintenance of the structures. This is in addition to the obvious benefits of early closure, recognised in [1], of greater safety, greater security against terrorist attack and a reduction in the worker dose burden transferred to future generations.

Various studies have been undertaken internationally to demonstrate retrieval of waste packages. In particular, Nirex has demonstrated the feasibility of using high-pressure water jets to retrieve ILW packages from disposal tunnels backfilled with NRVB [46]. More recently, SKB (the Swedish Nuclear Fuel and Waste Management Company) has demonstrated that waste canisters can be retrieved from a saturated bentonite buffer by slurrying it with a saline solution [47].

If provision is to be made for possible retrieval of the waste packages at some point in the future then it may be necessary to monitor the condition of the packages. A key technical issue here, therefore, is the need to better understand package longevity and the corresponding degradation mechanisms over a long period of storage. As part of the work underpinning the generic DSSC, further research into package longevity has been commissioned [48]. The aim of the work is to further investigate the time period over which package integrity is not unduly impaired by degradation processes that may occur under storage conditions prior to backfilling. This work will help inform decisions on whether package refurbishment or repackaging of wastes will be required if retrieval of packages were to be required after emplacement.

### 5.11.2 Decommissioning and closure strategy

At some stage in the future the GDF will be closed. Depending on the host rock of the selected site, this could potentially involve the backfilling of vaults, as well as backfilling of service tunnels, sealing of the underground openings and backfilling and closure of the access ways and drift or shafts. It is currently assumed that this would be implemented over a ten year period following emplacement of all wastes and subject to the provision of a satisfactory safety case and consultation with stakeholders. However this is unlikely to take place for at least another century. In practice, the duration of the emplacement period will be dependent on a number of factors, including decommissioning strategies for nuclear power stations and the provisions for handling wastes from any new build nuclear power programme. The point at which a GDF is closed will also be influenced by future decisions regarding retrievability, as discussed above.

Licence Condition 35 requires licensees to make adequate arrangements for decommissioning of facilities on a nuclear licensed site. All GDF surface facilities will eventually require to be decommissioned and this will be taken into account at all stages in the site life-cycle, starting at the planning and design stage. The extent to which underground facilities are decommissioned will depend on the value placed, in the future, on recovering the conventional (i.e. non-radioactive) materials used in those facilities. The decommissioning strategy for a GDF will therefore need to be closely linked to the lifetime emplacement strategy and be kept under review as the needs of the nuclear industry and the wishes of society regarding the future disposition of a GDF become more clearly defined.

Given the above and the long timescales involved, it is not possible to address decommissioning and closure operations in any detail at present. However, the need for future decommissioning will be kept under consideration throughout the design process.

Before delivery to a GDF, all radioactive wastes will be rendered passively safe through treatment and conditioning as appropriate and packaging in robust containers. Operations both above and below ground will be designed to maintain such containment. Therefore, it is intended that all facilities upstream of the disposal areas, will be operated with very low levels of surface contamination and airborne activity. The maintenance of a clean environment will greatly facilitate subsequent decommissioning and, for the surface facilities in particular, should allow much of the building and infrastructure materials and equipment to be recovered and released for re-use.

Ease of decontamination, and hence decommissioning operations, will be facilitated by design through, for example, the use of non-porous material such as stainless steel for process equipment and decontaminable coatings and finishes on surfaces where practicable. So far as possible, equipment will be designed so as to avoid potential contamination traps.

For the surface facilities, it is unlikely that novel decommissioning techniques beyond standard decontamination, dismantling and demolition methods will be required. Further, a considerable body of additional experience in the decommissioning of a wide range of nuclear facilities will have been accumulated by the time decommissioning at a GDF is due to commence. The practicability and desirability of decommissioning underground facilities will depend to a large extent on the characteristics of the site and its geology. However, none of the underground facilities are anticipated to require the development of novel techniques beyond established mining practices.

Overall, therefore, there is no reason to believe that decommissioning operations could not be carried out safely, maintaining worker doses below radiological dose targets. As the design and safety case is developed, consideration will be given to the ease of decommissioning so that the risks associated with decommissioning activities will be capable of being controlled to levels which are as low as reasonably practicable.

Closure will be preceded by backfilling and sealing of open vaults, service tunnels and access ways (including the drift or shafts). With the evaporite host rock, it is currently envisaged that no backfilling of the disposal vaults would be required although buffering of the packages with MgO is envisaged in our illustrative design [33]. For the higher strength and lower strength sedimentary host rocks, the crown spaces in the ILW/LLW and UILW vaults will be filled remotely from backfill galleries or distribution pipes which will be installed during construction of the vaults. The backfilling and closure operations will not involve any handling of waste packages and the radiological inventory will be restricted to the emplacement areas. Thus any operational hazards will be largely limited to the conventional hazards normally associated with large-scale cement-handling operations and will be controlled in accordance with accepted good practice and industry guidance.

During the backfilling period the rate of gas generation from the waste may increase in the short term due to heat generated by the cementation and curing processes. This would be managed through an appropriate backfilling strategy and ventilation. The potential impact on gaseous emissions to atmosphere is discussed in the Operational Environmental Safety Assessment Report [40].

## 5.12 Construction and non-radiological safety

### 5.12.1 Introduction

The construction of a GDF will be a major undertaking and we will draw upon the very large body of available experience from the mining and tunnelling industries, and will employ proven technologies and methods. In undertaking the construction of a GDF we would build the requirements of the Construction Design and Management Regulations (CDM), into our design and management processes. This will help to improve planning and management of the construction from the outset and to ensure that RWMD, designers and contractors have clearly defined and understood roles and responsibilities. The way CDM regulation may impact on the design and construction of a GDF is discussed further in [6]. In addition there is a large body of regulatory requirements, some of which are specific to mining and tunnelling activities. Although a GDF may not meet the legal definition of a mine, we will take due account of the mining regulations as representing best practice as part of our design and construction work. The relevant regulations are presented in [6]. Reference [6] also provides a discussion of the implications of the applicable legislative requirements and the available regulatory guidance.

We have examined the potential hazards associated with the construction and commissioning of a GDF, together with the approach and management strategies and controls to be adopted prior to and throughout construction and commissioning. This work is presented in [6] and briefly summarised in this section.

Of particular interest is how construction activities interact with research and site characterisation activities and, subsequently, with waste emplacement activities in already constructed and commissioned vaults. Whilst the construction will follow industry standard safety procedures, the interactions between different teams will also need to be addressed.

### 5.12.2 Construction methods

The construction methods employed for the various excavations will depend on the geology of the chosen site, in particular: the strength of the rock; the fractured condition of the rock mass and the groundwater inflow.

For higher strength rock, it is anticipated that the majority of the excavation work will be carried out using traditional drill and blast techniques. A purpose-built tunnel boring machine (TBM) could potentially be used for the construction of the drift.

For the lower strength sedimentary rock, TBM (for long tunnels), road-headers or drill and blast methods could all potentially be used.

In the case of the evaporite rock, it is anticipated that drill and blast methods would be used for shaft sinking, while continuous miner or road-header machines would be used for the remainder of the excavations.

### 5.12.3 Construction hazards

Preliminary Hazard Identification studies have been held to identify the potential hazards associated with the GDF construction phase [6]. A fault schedule has been generated from the output of these studies and is also presented in [6]. The fault schedule highlights the principal hazards and safeguards associated with construction. The services and infrastructure required to support construction will be important in maintaining safe working conditions. The fault schedule therefore includes the hazards associated with the use of and potential damage to services and infrastructure equipment.

A detailed discussion of the potential hazards identified and the means by which these can be prevented or mitigated against is given in [6]. A summary of the main hazards is provided below.

**Fire and explosion hazards**

Sources of fire during underground excavations include sparks from electrical equipment or static discharge; hot working (e.g. welding); spillage of fuel or oil from vehicles and equipment (possibly as a result of impact damage) and the presence of flammable gases (e.g. methane).

All electrical equipment used in construction would be appropriately rated and earthed and would be subject to regular inspection. Hot works would be controlled by permit and would only be carried out by trained and experienced personnel. Consideration will be given to the minimisation of fire hazards from an early stage in construction planning, in particular in the selection of vehicles and other equipment required to be taken underground. Equipment liable to generate incendive sparking, such as any TBM, road headers or miners, would be fitted with automatic and/or manually generated fire suppression devices where appropriate. It is noted that incendive sparking may be more of an issue for the lower strength sedimentary rock since this tends to be associated with the smaller TBMs. Low flammability hydraulic oils and fluids would be used as far as possible.

A manned, fully equipped fire station would be provided at the surface while local fire suppression equipment would be provided underground and personnel trained in its use. Underground refuges, designed to modern standards, would be installed, with respirators and other personnel protective equipment provided at intermediate points as required. Fire detection and alarm systems would be installed as appropriate.

The GDF is unlikely to be sited near hydrocarbon-bearing beds which are most likely to generate flammable gases such as methane. However, gas can permeate into non-hydrocarbon strata. The key protection here will be the site investigations, coupled with advance drilling and characterisation local to the workface.

There are particular hazards associated with the use of drill and blast methods. It is anticipated that drill and blast would be required for shaft construction in all three concepts. Otherwise, drill and blast is most commonly used for excavations in higher strength rock, although it could also be used in the lower strength sedimentary rock.

An obvious requirement is the need to control the storage of explosives and detonators on site. A conventional explosives store would be provided at the surface to ensure that the explosives required for the underground excavation operations can be held safely and securely. In accordance with standard practice in the mining and quarrying industry, there would be separate stores for explosives and detonators. The locations of the stores would be required to comply with statutory legislation regarding proximity to inhabited buildings. The need to maintain a safe distance from the surface active facilities would also be taken into account. Handling and use of explosives underground would also be strictly controlled and blasting would only be carried out by suitably qualified and licensed personnel. With modern explosives, the risk of an uncontrolled explosion is, in any case, very low. Access to blasting areas would be restricted to essential personnel only. Controls would be put in place to ensure that waste explosives are destroyed correctly and that non-detonated holes are washed out and detonators recovered for correct disposal.

**Vibration hazards**

In the higher strength rock, blasting operations will be required in neighbouring areas, particularly in the HLW/SF depositions areas where numerous deposition tunnels are required with supporting transport and service tunnels. It will be important to ensure that all personnel are evacuated from affected areas prior to blasting. However, blast vibrations could potentially be felt at a considerable distance away from the source including the ILW emplacement operational areas. (Construction of the HLW/SF disposal areas will not begin until after ILW emplacement operations have commenced.) An associated hazard is the potential for sympathetic detonation or for blasting to affect the integrity of adjacent structures. Vibration from blasting operations at shallow depths can potentially result in a nuisance to the public as well.

These hazards will be controlled by careful design of the blasting patterns and detonation sequences, supported by prior mapping and inspection of the working area. Nuisance off-site can be minimised by appropriate scheduling. Coordination of work plans and good communication between the construction and emplacement teams and the specialist mapping engineers will be vital here.

**Exposure to noxious fumes, gases and dusts**

The underground excavation work will inevitably generate considerable amounts of dusts. Noxious fumes will also be generated from vehicle exhausts and blasting operations. This can lead to poor breathing conditions and (in the case of dust) impair visibility. Radioactive radon gas may also escape from the host rock

The primary means of control here will be adequate ventilation, supported by the provision of fresh air bases at refuges and respiratory protection equipment. However use of respiratory protection would only be acceptable for short periods for specific operations. It would not be considered as a routine means of protection. Local dust suppression may be employed at the working face.

The need to minimise exhaust fumes would be one of the considerations in selecting vehicles and other heavy equipment, e.g. the use of electrically driven vehicles and the use of conveyors for transporting spoil away from the working face. Diesel equipment would be fitted with fume diluters or catalytic/water scrubbers as required.

Personnel would be evacuated from the working face where blasting is taking place and would not be permitted to return until the fumes had dispersed. Where there are several working faces operating at the same time, the workers may evacuate to refuge areas (which would be positively ventilated) rather than returning to the surface.

Radon monitoring would be undertaken where required.

The most difficult area to ventilate during the construction phase is likely to be the tunnel complex in the HLW/SF disposal areas. Many of the tunnels will be closed at one end initially and may require forced ventilation. Construction of the service tunnels will be planned so as to permit the establishment of complete ventilation circuits as early as practicable.

The illustrative concept example for higher strength rock also includes the provision of 1m diameter vertical deposition holes to contain the HLW/SF/Pu/HEU disposal canisters. The deposition holes will require to be inspected following drilling and again prior to commissioning to ensure that the dimensions are within the specified tolerance and that the holes are clear of debris or liquids (such as drilling fluids, groundwater) which could prevent effective emplacement of the disposal canisters or lead to subsequent degradation of the canisters. Given the potential for dense gases to accumulate in the deposition holes, purging would be carried out prior to man-entry as appropriate.

The potential for general construction hazards such as dropped loads, trips and falls, falls from height and vehicle collisions will also be present and will require to be controlled. These are discussed in [6]. Control of environmental hazards such as groundwater management and handling of contaminated liquids such as drilling fluids are also discussed.

### 5.12.4 Safety management and human factors

There will be a number of different teams present during the first phase of construction and commissioning. Subsequently the construction of new vaults will be undertaken in parallel with package emplacement operations. Hence, those present underground will include site characterisation personnel, construction personnel, emplacement personnel and specialists such as geotechnical engineers. There is evidence from incidents that have occurred in similar multi-disciplinary engineering projects, which a lack of appreciation of respective roles and communication between teams can lead to accidents and even fatalities [6]. It will be vital, therefore, to establish a clear and effective management organisation and promote the right behaviours throughout the design, construction and operation of the facility, in order to ensure that risks are reduced as low as reasonably practicable.

A behaviour-based approach to the management of safety is being developed [6,49] which pays particular attention to the organisational risk reduction processes that could be implemented to optimise the safety and welfare of the workers, whilst constructing and operating the GDF. In [49] the range of interactions between different teams is considered and a bespoke management system model is developed, detailing the types of organisational arrangements that would be implemented to support and reinforce a safe and productive working environment.

#### Interactions with radiological safety

The continuing integrity of a GDF against potential hazards such as structural failure and flooding will be reliant on effective and appropriate construction and ongoing monitoring. Access to certain areas, such as the disposal vaults containing shielded packages, will be very limited once emplacement operations are underway so any remedial work to the structure could potentially be both costly and technically challenging.

Effective protection against groundwater ingress will also be important, both for the safety of workers during construction and to prevent flooding and degradation of the structure during the subsequent emplacement and monitoring phases. This will be a key issue in any evaporite host rock since undetected groundwater ingress, following construction, could potentially ‘dissolve’ affected parts of the structure.

The choice of construction methods is normally made by the construction contractor. Once the construction methods have been determined, it will be necessary to ensure ongoing dialogue with the design team regarding the prevention and mitigation of risks.

The construction will be informed by extensive prior site characterisation studies. These will require to be supplemented by ongoing monitoring and characterisation at the working faces as construction proceeds. Good communication between the geotechnical specialists carrying out the characterisation and the engineers managing the construction team will therefore be vital and this will be reflected in the management arrangements.

Typically the key personnel would be the shift engineers at the working face. They would be responsible for selecting the appropriate support solutions at the working face based on the information supplied to them on the characteristics of the rock mass, adding any additional support they consider necessary to deal with local instabilities. Prior to construction, planned responses for dealing with unanticipated ground conditions would have been put in place. The shift engineer would be responsible for selecting and initiating the appropriate response should a fault be revealed.

With overall responsibility for the safety of the construction workers, the shift engineer would also be responsible for safety during blasting operations, ensuring that workers are evacuated from affected areas and that blasting fumes are sufficiently dispersed before work recommences. We will work closely with the construction team to ensure that suitably qualified and experienced personnel will be appointed for monitoring and recording of air quality at the working faces.

This introduces another area of interaction with the emplacement side, since noise and vibration from blasting operations could be experienced by operators involved in emplacement operations. In addition to the ongoing construction of the ILW/LLW disposal vaults, construction of the HLW disposal areas will not commence until approximately thirty years after the start of ILW/LLW emplacement operations. As with ILW/LLW, there will be a need for subsequent construction of further HLW deposition areas, on an ‘as required’ basis.

Hence, there will need to be effective means of conveying information from the construction side to the emplacement side (and vice versa) such that the emplacement teams and other construction teams are given warning of, for example blasting operations.

We will ensure that our safety management system provides for adequate integration of the management of the construction operations with the management of active operations and for effective communication between the two sets of operators.

We believe the adoption of a common management system would also facilitate a common approach to conventional safety issues through the entire facility, ensuring that adequate attention is given to non-radiological hazards such as falls from height, exposure to chemical fumes etc. in both active and non-active areas.

This is discussed further in [6].

## 6 Conclusions and forward programme

This report presents the key information and findings from the assessment reports that comprise the generic OSC. The primary supporting documents are the Safety Case Production and Management Report [10] and the operational safety assessment reports: Construction and Non-radiological Safety Assessment [6]; Normal Operations Operator Dose Assessment [7]; Accident Safety Assessment [8] and Criticality Safety Assessment [9].

In this report we have outlined the safety strategy in terms of the integrated approach to design and safety case development.

We have summarised the key principles, procedures and methodologies we are developing for producing safety assessments and presented an example assessment, based on the illustrative concept examples we have developed for a GDF, in three different host rocks.

In the example assessment we have considered both normal operations and potential fault conditions. In the normal operations assessment we have estimated worker doses due to routine operations at a number of facilities on the surface and in the underground disposal areas. (Routine off-site discharges and disposals during normal operations are discussed in the OESA [40].) In order to do this, we have had to make a number of assumptions regarding facility layouts and operator work locations. Despite a conservative approach, the preliminary results indicate that we will be able to meet both the legal limits and our corporate dose targets.

In the accident safety assessment, we have identified the most safety significant faults and considered the consequences of these. One of the primary findings of the analysis is the key importance of the safety benefit provided by the robust transport containers in which waste packages containing the higher inventory wastes are transported to the site. In all the illustrative concept examples, the waste packages remain in their transport containers from arrival at the site until they are transferred to remote handling facilities underground. The transport containers are required, under transport regulations, to be able to withstand significant fire and impact challenges and this protection effectively eliminates the consequences which could arise from a large number of potential fault scenarios.

The assessment also clearly illustrates the importance of the robust design of the disposal canisters for HLW/SF and Pu/HEU, both in relation to their resistance to fire and impact damage and their radiation shielding properties.

The results of the deterministic (DBA) fault analysis show that for some faults we do not meet our corporate safety criteria [26]. This is due in part to an overly conservative approach to the DBA. This pessimism has two main sources. First the probability used to set the consequence acceptability threshold is taken as the highest for all consequence bands and second, the calculated consequences themselves are pessimistic because conservative release fractions (RFs) from impact and fire are used in the calculations.

We will establish a set of conservative initiating fault frequencies for the DBA and then we will revise our analysis and, where appropriate, take a more realistic (but still conservative) approach to the fault assessment using less pessimistic RFs. In revising the assessment we will also investigate whether we can reasonably take account of any additional engineered safety features in the design.

Notwithstanding the preliminary and illustrative nature of the assessment, we believe that the findings provide confidence that it will be possible to develop a GDF design which will be capable of meeting modern safety standards for nuclear facilities in compliance with nuclear site licence conditions. This is confirmed by the findings of the high-level probabilistic safety assessment undertaken which, even taking account of the most radiologically challenging waste streams that the facility will be required to accommodate, indicate that our corporate risk targets would be met for both workers and members of the public. Our conclusions are supported by the fact that our designs and planned operations draw on the UK's 60 years of experience of safely processing, handling and storing waste in surface facilities throughout the UK.

In addition, we have begun consideration of safety during construction of a GDF, with preliminary hazard identification studies being undertaken. Whilst a large number of potential hazards associated with the construction of a GDF have been identified, there are no hazards identified for which there are no means of protecting the workforce or the public from unacceptable consequences. This, together with the fact that the construction of a GDF will utilise the very large body of available experience from the mining and tunnelling industries, and will employ proven technologies and methods, gives us confidence that construction risks can be demonstrated to be acceptably low.

## 6.1 Forward programme

During the current period of preparatory work, we will continue our research and development studies to extend our present understanding of individual aspects of geological disposal. The work we are planning to undertake in support of our safety assessments includes:

- Review of initiating fault frequencies for DBA and revision of DBA to remove any unnecessary conservatism
- Investigation of waste package release fractions for a range of fire and impact scenarios
- Further investigation of the time period over which package integrity is not unduly impaired by degradation processes that may occur under storage conditions prior to backfilling.
- Investigation into inventories of gaseous radio-elements in different waste types and the effect of temperature and chemical form on their potential release rates
- Investigation into the potential for exposure under normal operating conditions to airborne activity from waste packages and naturally occurring gases
- Review of current assumptions regarding the release of particulate generated in an impact scenario and the fraction of material expected to be respirable
- Review of the parameters used for calculation of public doses
- Identification of definitive bounding waste streams

We believe these studies will help to reduce the uncertainties in our current methodologies and permit a reduction in the degree of conservatism they currently contain.

Once we start to develop site-specific designs, we will undertake detailed hazard identification studies which will enable us to refine our fault schedule and future safety assessments.

## Glossary

*Activity*

The number of atoms of a radioactive substance which decay (radioactive decay) by nuclear disintegration each second. The SI unit of activity is the Becquerel (Bq).

*Advanced Gas-cooled Reactor (AGR)*

The reactor type used in the UK's second generation nuclear power plants.

*Alpha activity*

Alpha activity takes the form of particles (helium nuclei) ejected from a decaying (radioactive) atom. Alpha particles cause ionisations in biological tissue which may lead to damage. The particles have a very short range in air (typically about 5 cm) and alpha particles present in materials that are outside of the body are prevented from doing biological damage by the superficial dead skin cells, but become significant if inhaled or swallowed.

*Backfilling*

The refilling of the excavated portions of a disposal facility after emplacement of the waste.

*Barrier*

A physical obstruction that prevents or inhibits the movement of radionuclides.

*Baseline Inventory*

An estimate of the higher activity radioactive waste and other materials that could, possibly, come to be regarded as wastes that might need to be managed in the future through geological disposal drawn from the UK Radioactive Waste Inventory.

*Becquerel (Bq)*

The standard international unit of radioactivity equal to one radioactive decay per second. Becquerels are abbreviated to Bq. Multiples of Becquerels commonly used to define radioactive waste activity are: kilobecquerels (kBq) equal to 1 thousand Bq; megabecquerels (MBq) equal to 1 million Bq; gigabecquerels (GBq) equal to 1 thousand million Bq.

*Bentonite*

A highly sorbing clay material used as a backfill in certain disposal concepts.

*Beta activity*

Beta activity takes the form of particles (electrons) emitted during radioactive decay from the nucleus of an atom. Beta particles cause ionisations in biological tissue which may lead to damage. Most beta particles can pass through the skin and penetrate the body, but a few millimetres of light materials, such as aluminium, will generally shield against them.

*Buffer*

The buffer is an engineered barrier that protects the waste package and limits migration of radionuclides following waste package failure.

*Closure*

Technical and administrative actions to put a disposal facility in its intended final state after the completion of waste emplacement.

*Committee on Radioactive Waste Management (CoRWM)*

CoRWM was set up in 2003 to provide independent advice to the UK Government on the long-term management of the UK's solid higher activity radioactive waste. In October

2007, CoRWM was reconstituted with revised Terms of Reference and new membership. The Committee will provide independent scrutiny and advice to UK Government and devolved administration Ministers on the long-term radioactive waste management programme, including storage and disposal. Further information available at http://www.corwm.org.uk

*Community Siting Partnership*

A partnership of local community interests that will work with the NDA's delivery organisation and with other relevant interested parties to ensure questions and concerns of potential Host Communities and its Wider Local Interests are addressed and resolved so far as reasonably practicable and to advise Decision Making Bodies at each stage of the process.

*Conditioning*

Treatment of a radioactive waste material to create, or assist in the creation of, a wasteform that has passive safety.

*Conditions for Acceptance*

Quantitative and/or qualitative criteria, specified by the operator of a waste handling facility, which define the conditions under which waste will be accepted into that facility. (In the case of a disposal facility such criteria are usually referred to as Waste Acceptance Criteria).

*Containment*

A feature of a geological disposal facility component that contributes to safety. Some engineered barriers provide a significant period of containment of radionuclides by physically confining them to prevent their release into the host rock. Other barriers may also contribute to containment by providing sufficient travel time to allow decay of some radionuclides and their daughters to negligible levels.

*Criticality*

Criticality is a state in which a quantity of fissile material can maintain a self-sustaining neutron chain reaction. Criticality requires that a sufficiently large quantity of fissile material (a critical mass) be assembled into a geometry that can sustain a chain reaction; unless both of these requirements are met, no chain reaction can take place – the system is said to be sub-critical.

*Decision-making body*

Local Government will have decision-making authority for their host community in respect of continued participation at key stages in the siting process, or exercising a Right of Withdrawal; the local acceptability of proposals for Community Benefits Packages; the local acceptability of sites proposed for surface investigations; and whether potential retrievability of wastes has been adequately considered. There are different local authority structures in different parts of the UK. For example, in England local authorities include district councils, county councils, metropolitan district councils and London Boroughs whereas in Wales, local authorities are unitary. Such a body is termed a 'Decision-Making Body'.

*Decommissioning*

The process whereby a nuclear facility, at the end of its economic life, is taken permanently out of service and its site made available for other purposes.

*Depleted Uranium (DU)*

Uranium in which the proportion of U-235 is less than ~0.7%.

*Deposition Hole*

Vertical hole within a deposition tunnel in which a HLW, SF, Pu or HEU canister is placed for disposal.

*Disposal*

In the context of solid waste, disposal is the emplacement of waste in a suitable facility without intent to retrieve it at a later date; retrieval may be possible but, if intended, the appropriate term is storage.

*Disposal canister*

A term used to describe the assembly of certain waste types (e.g. HLW, SF, plutonium, HEU) within a metal container, as prepared for disposal.

*Disposal unit*

A waste package, or group of waste packages, which is handled as a single unit for the purposes of transport and/or disposal.

*Dose, Dose rate*

Dose is a measure of exposure to radiation and can be taken to mean effective dose equivalent unless stated otherwise. Similarly, 'Dose rate' would mean the effective dose equivalent per unit time. The SI unit of effective dose equivalent is the sievert (Sv), and typical units of dose rate are sievert/hour (Svh-1) and sievert/year (Svy-1).

*Dose equivalent*

Dose equivalent takes into account not only the energy deposited in body tissue by radioactivity (either external or internal) but also the different biological effectiveness of the various forms of radiation in causing harm to body tissues. The SI unit of dose equivalent is the sievert (Sv).

*Effective dose equivalent*

In addition to dose equivalent taking into account the biological effectiveness of various forms of radiation, effective dose equivalent takes into account the differing sensitivities of various body tissues. Effective dose equivalent thus aims to reflect the risk to health for the irradiated person, regardless of the widely different dose equivalents that might be received by the various organs. This is a useful concept for comparisons between the risks from various radiation exposure pathways.

*Enrichment (Uranium)*

The proportion (usually expressed as a % of the total mass) of U-235 in uranium.

*Evaporite*

The generic term for a geological environment created by the evaporation of water from a salt bearing solution to form a solid structure.

*Fissile material*

Fissile material is a material that undergoes fission under neutron irradiation. For regulatory purposes material containing any of the following nuclides is considered to be 'fissile': U-233, U-235, Pu-239, Pu-241.

*Gamma activity*

An electromagnetic radiation similar in some respects to visible light, but with higher energy. Gamma rays cause ionisations in biological tissue which may lead to damage. Gamma rays are very penetrating and are attenuated only by shields of dense metal or concrete, perhaps some metres thick, depending on their energy. Their emission during radioactive decay is usually accompanied by particle emission (beta or alpha activity).

*Geological disposal*

A long-term management option involving the emplacement of radioactive waste in an engineered underground geological disposal facility or repository, where the geology (rock structure) provides a barrier against the escape of radioactivity and there is no intention to retrieve the waste once the facility is closed.

*Higher strength rock*

Typically crystalline igneous and metamorphic rocks or geologically older sedimentary rocks where any fluid movement is predominantly through discontinuities.

*High Enriched Uranium (HEU)*

Uranium in which the proportion of U-235 is greater than ~20%.

*High Level Waste (HLW)*

Radioactive wastes in which the temperature may rise significantly as a result of their radioactivity, so this factor has to be taken into account in the design of storage or disposal facilities.

*Immobilisation*

A process by which the potential for the migration or dispersion of the radioactivity present in a material is reduced. This is often achieved by converting the material to a monolithic form that confers passive safety to the material.

*Industrial Package*

A category of transport package defined by the IAEA Transport Regulations. Industrial Packages are restricted to the carriage of Low Specific Activity material and/or Surface Contaminated Objects.

*Intermediate Level Waste (ILW)*

Radioactive wastes exceeding the upper activity boundaries for LLW but which do not need heat to be taken into account in the design of storage or disposal facilities.

*Letter of Compliance (LoC)*

A document, prepared by RWMD, that indicates to a waste packager that a proposed waste package is compliant with the relevant packaging criteria and disposal safety assessments, and is therefore deemed to be compatible with disposal in a GDF.

*Low Enriched Uranium (LEU)*

Uranium in which the proportion of U-235 is greater than ~0.7% but less than ~20%.

*Low Level Waste (LLW)*

LLW is defined as “radioactive waste having a radioactive content not exceeding 4 gigabecquerels per tonne (GBq/te) of alpha or 12 GBq/te of beta/gamma activity”.

*Low Level Waste Repository*

The UK national facility for the near surface disposal of solid LLW, located near to the village of Drigg in Cumbria.

*Low Specific Activity material*

A material classification defined by the IAEA Transport Regulations as ‘Radioactive material which by its nature has a limited specific activity (i.e. activity per unit mass of material), or radioactive material for which limits of estimated average specific activity apply.’

*Managing Radioactive Waste Safely (MRWS)*

A phrase covering the whole process of public consultation, work by CoRWM, and subsequent actions by the UK Government, to identify and implement the option, or combination of options, for the long term management of the UK's higher activity radioactive waste.

*Natural Uranium*

Uranium in which the proportion of U-235 is ~0.7%

*Nirex (UK Nirex Ltd)*

An organisation previously owned jointly by Defra and the DTI. Its objectives were, in support of UK Government policy, to develop and advise on safe, environmentally sound and publicly acceptable options for the long-term management of radioactive materials in the United Kingdom. The Government's response to CoRWM in October 2006 initiated the incorporation of Nirex functions into the NDA, a process which was completed in March 2007.

*Nuclear Decommissioning Authority (NDA)*

The NDA is the implementing organisation, responsible for planning and delivering the GDF. The NDA was set up on 1 April 2005, under the Energy Act 2004. It is a non-departmental public body with designated responsibility for managing the liabilities at specific sites. These sites are operated under contract by site licensee companies (initially British Nuclear Group Sellafield Limited, Magnox Electric Limited, Springfields Fuels Limited and UK Atomic Energy Authority). The NDA has a statutory requirement under the Energy Act 2004, to publish and consult on its Strategy and Annual Plans, which have to be agreed by the Secretary of State (currently the Secretary of State for Trade and Industry) and Scottish Ministers.

*Nuclear licensed site*

Any site which is the subject of a licence granted by the Nuclear Installations Inspectorate (part of HSE) under the Nuclear Installations Act 1965. Nuclear licensed sites include nuclear power stations, nuclear fuel production and reprocessing sites, sites undertaking storage of and/or research into nuclear materials, and major plant producing radioisotopes.

*Passive safety*

A passively safe wasteform is one in which the waste is chemically and physically stable, and is stored in a manner that minimises the need for safety mechanisms, maintenance, monitoring and human intervention, and that facilitates retrieval for final disposal.

*Plutonium (Pu)*

A radioactive element occurring in very small quantities in uranium ores but mainly produced artificially, including for use in nuclear fuel, by neutron bombardment of uranium.

*Pressurised Water Reactor (PWR)*

Reactor type using ordinary water under high pressure as coolant and neutron moderator. PWRs are widely used throughout the world for electricity generation. The Sizewell B reactor in Suffolk is of this design.

*Radioactive material*

Material designated in national law or by a regulatory body as being subject to regulatory control because of its radioactivity.

*Radioactive waste*

Any material contaminated by or incorporating radioactivity above certain thresholds defined in legislation, and for which no further use is envisaged, is known as radioactive waste.

*Radioactive Waste Management Directorate (RWMD)*

The NDA Directorate established to design and build an effective delivery organisation to implement a safe, sustainable, publicly acceptable geological disposal programme. It is envisaged that this directorate will become a wholly owned subsidiary company of the NDA. Ultimately, it will evolve under the NDA into the organisation responsible for the delivery of the GDF. Ownership of this organisation can then be opened up to competition, in due course, in line with other NDA sites.

*Radioactivity*

Atoms undergoing spontaneous random disintegration, usually accompanied by the emission of radiation.

*Radionuclide*

A radioactive form of an element, for example carbon-14 or caesium-137.

*Reprocessing*

A physical or chemical separation operation, the purpose of which is to extract uranium or plutonium for re-use from spent fuel.

*Shielded waste package*

A shielded waste package is one that either has in-built shielding or contains low activity materials, and thus may be handled by conventional techniques. In most cases, shielded waste packages are also designed to qualify as transport packages in their own right.

*Shielding*

Shielding is the protective use of materials to reduce the dose rate outside of the shielding material. The amount of shielding required to ensure that the dose rate is as low as reasonably practicable (ALARP) will therefore depend on the type of radiation, the activity of the source, and on the dose rate that is acceptable outside the shielding material.

*Spent fuel (SF)*

Used fuel assemblies removed from a nuclear power plant reactor after several years use and treated either as radioactive waste or via reprocessing as a source of further fuel.

*Stillage*

A metal frame designed to hold four 500 litre Drums waste packages so that they can be handled, stacked and transported as a single disposal unit.

*Surface Contaminated Object*

A material classification defined by the Transport Regulations as 'A solid object which is not itself radioactive but which has radioactive material distributed on its surfaces.'

*Transport container*

A reusable container into which waste packages are placed for transport, the whole then qualifying as a transport package under the Transport Regulations.

*Transport package*

As defined in the IAEA Transport Regulations: 'the complete assembly of the radioactive material and its outer packaging, as presented for transport.'

*UK Radioactive Waste Inventory (UKRWI)*

A compilation of data on UK radioactive waste holdings, produced about every three years. The latest version, for a holding date of 1 April 2007, was published in June 2008. It is produced by Defra and the NDA. It is the latest public record of information on the sources, quantities and properties of LLW, ILW and HLW in the UK. It comprises of a number of reports and additional detailed information on the quantities and properties of radioactive wastes in the UK that existed at 1 April 2007 and those that were projected to arise after that date.

*Unshielded waste package*

An unshielded waste package is one that, owing either to radiation levels or containment requirements, requires remote handling and must be transported in a reusable transport container (the container and contents then forming a Type B transport package).

*Uranium (U)*

A heavy, naturally occurring and weakly radioactive element, commercially extracted from uranium ores. By nuclear fission (the nucleus splitting into two or more nuclei and releasing energy) it is used as a fuel in nuclear reactors to generate heat.

Uranium is often categorised by way of the proportion of the radionuclide uranium U-235 it contains (see natural uranium, depleted uranium, low enriched uranium and high enriched uranium).

*Wasteform*

The waste in the physical and chemical form in which it will be disposed of, including any conditioning media and container furniture (i.e. in-drum mixing devices, dewatering tubes etc) but not including the waste container itself or any added inactive capping material.

*Waste packager*

An organisation responsible for the packaging of radioactive waste in a form suitable for transport and disposal.

*Waste producer*

An organisation responsible for the creation and/or storage of radioactive waste in an unconditioned form.

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## Appendix A Legislation applicable to the construction and operation of the GDF

The UK has a strong and effective regulatory regime for the management of radioactive waste, including storage and transport. Implementation of the geological disposal facility programme by the NDA will comply fully with relevant UK and international legislation and conventions, including:

- all relevant Euratom Treaty requirements as transposed into UK law, including Council Directive 96/29/Euratom laying down basic safety standards for the protection of the health of workers and the general public against the dangers of ionising radiation [1] (the Basic Safety Standards Directive)
- all relevant legislation, including the Health and Safety at Work etc. Act 1974 (HSWA74) [2], the Nuclear Installations Act 1965 (NIA65)[3], the Environmental Permitting (England and Wales) Regulations 2010 [4], the Carriage of Dangerous Goods and Use of Transportable Pressure Equipment Regulations 2007 [5] and the Nuclear Industries Security Regulations 2003 [6]
- the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management, and the Convention on Physical Protection of Nuclear Material
- the principles of radiological protection established by the International Commission on Radiological Protection (ICRP) as reflected in European Union and UK legislation and standards, the latter based on independent advice from bodies such as the Health Protection Agency (HPA) and the Committee on Medical Aspects of Radiation in the Environment (COMARE).

In the United Kingdom, the main legislation governing the safety of nuclear installations is the Health and Safety at Work Act 1974 (HSWA) and the associated relevant statutory provisions of the Nuclear Installations Act 1965 (NIA65) and the Ionising Radiation Regulations (1999).

The **Health and Safety at Work Act 1974** deals with hazards arising from all work activities, including nuclear hazards, and places 'general duties' on employers for the safe design and operation of their plant so as to ensure the health and safety of their employees and of other persons. All civil nuclear installations, licensed or not, are subject to these general provisions including other relevant statutory provisions, such as the Ionising Radiations Regulations 1999.

The **Nuclear Installations Act, 1965 (as amended 1969)** requires that, before a nuclear installation can be constructed or operated on any site for specified activities, a nuclear site licence must be granted to a licensee by the regulatory body and the licence be currently in force. This Act also enables conditions to be attached to a site licence in the interests of nuclear safety both in normal circumstances and in emergencies. These conditions may be altered at any time to ensure that adequate standards are developed, achieved and maintained by the operator throughout his "period of responsibility". The use of this approach to regulate and control nuclear safety by means of its power under the site licence provides the NII with an enforcement tool, which is both powerful and flexible enough to match the degree of risk involved. The activities for which a licence is required include the processing and long-term storage of radioactive waste. Therefore, it is anticipated that any site upon which it is intended to develop a geological disposal facility will require a Nuclear Site Licence.

Licence conditions cover all aspects of nuclear safety relating to the development of the facility and provide for a series of construction and operational hold points e.g. consent to start construction or excavation, consent to start commissioning, etc. Before work can proceed beyond a hold point, the Health and Safety Executive will need to be satisfied that the proposed activity following the hold point is backed by a satisfactory safety case submission. Safety case submissions are documents required to be produced by applicants for nuclear site licenses and by existing licensees under their nuclear site licensing conditions to allow the safety regulator (NII) to assess, and thus ensure, the safety of their proposed operation practices and arrangements. The process by which HSE considers applications for nuclear site licences is described in the Licensing of Nuclear Installations [7] The technical principles, which NII uses to judge a licensee's safety case, are expressed in Safety Assessment Principles for Nuclear Facilities [8].

The radiation doses to workers and members of the public are subject to the requirements of the **Ionising Radiation Regulations (IRRs).** They set upper limits on radiation doses but irrespective of such limits also require that an employer shall take all necessary steps to restrict as far as reasonably practicable the extent to which people are exposed to ionising radiation.

While the HSE (through the NII) will be responsible for regulating the construction and operation of a geological disposal facility, the environmental agencies are responsible for regulating the disposal of radioactive waste under the **Environmental Permitting Regulations 2010 (EPR10)**. We (the NDA's delivery organisation) will be required to apply to the appropriate environmental regulator for a permit under the Radioactive Substances Regulation (RSR) of EPR10. As well as the final disposal of the radioactive waste inventory, permits will be required for any discharge of radioactivity to the environment during operation of the facility.

Before the environmental regulator grants any permit, the European Commission (EC), under Article 37 of the Euratom Treaty [9], will also need to be satisfied that other countries will not be adversely affected by the proposed disposal facility. Within any permit for radioactive waste disposal, the environmental regulator has the right to impose additional controls on us to ensure protection of the environment from a non-radiological perspective.

The development of a geological disposal facility will be subject to staged authorisation by the environmental regulator. Following careful consideration of responses to the MRWS consultation, Government is looking to amend the legislative powers available to the Environment Agency to enable it to undertake a staged authorisation process more effectively.

Civil nuclear installations must have a site-specific security plan approved by the OCNS as required by the **Nuclear Industries Security Regulations.** The security plan must provide details on site security management, policing and guarding, and to describe in detail the site security measures and arrangements for managing and reporting incidents. OCNS approval of carriers and transport plans will also be required where movement of nuclear material to the facility is involved. It is intended that the OCNS will ensure that security measures are included in plans for the construction of any new facility from the outset. This will enable regulators to make an early judgement on the most appropriate measures for any construction site and help ensure that security is ingrained into practices at a site from day one.

**Nuclear safeguards** are international measures that assure individual states comply with their international obligations not to use civil nuclear materials (plutonium, uranium and thorium) for nuclear explosives purposes. The International Atomic Energy Agency (IAEA) can choose which civil nuclear material in UK facilities it verifies, but the EC must apply safeguards to all such material according to the requirements of Chapter 7 of the Euratom Treaty [9] and Euratom Safeguards Regulation 302/2005 [10].

Waste will need to be transported safely from interim stores to the site of the geological disposal facility. The requirements for the safe transport of radioactive material by road, rail and sea stem from international agreements and European Directives. These requirements have been implemented in UK legislation setting out what types of transport package are allowed, how much radioactivity they are allowed to contain, and how they should perform against specified tests. Approval from the transport safety regulator is required for certain package designs and the competent authority in the UK is the Department for Transport who issues such approvals.

The legislation governing the transport of dangerous goods by road and rail, which include radioactive waste, in the UK is the **Carriage of Dangerous Goods and Use of Transportable Pressure Equipment Regulations 2007** [11]. These national regulations are derived from the international regulation of the IAEA Transport Regulations [12]. These form the basis for the national and international transport of radioactive materials and apply to the transport of radioactive material by all modes on land, water or in the air. For the PGRC, as the concept exists at this time, road and rail are seen as the principle transport routes to the GDF but other modes of transport such as sea are not discounted for which specific legislation also applies.

For nuclear licensed sites, **Radiation (Emergency Preparedness and Public Information) Regulations (REPPIR)** [13] establishes a framework of emergency preparedness measures to ensure that the population local to the site is:

- informed and prepared, in advance, about what to do in the unlikely event of a radiation emergency occurring; and,
- provided with information if a radiation emergency actually occurs.

The NDA will be developing emergency arrangements and plans for the GDF for dealing with any reasonably foreseeable radiation emergency. These plans will provide prior information to the population around the site and will detail the arrangements to be put in place in the unlikely event of such emergencies. The Regulations also place duties on the local authority in whose area the site is based, to prepare (and if necessary, implement) an off-site emergency plan for dealing with the consequences of any reasonably foreseeable radiation emergency in an area determined by HSE. The local authority is also required to ensure that relevant information is supplied to the affected population in the event that a radiation emergency should occur. REPPIR is enforced by HSE under the Health and Safety at Work Act.

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## Appendix B PSA results

The following tables proivde a comparison of the PSA results from our example assessment against our criteria. Results are given for three separate emplacement periods namely:

- Period 1 2040 – 2090
- Period 2 2080 – 2120
- Period 3 2120 - 2140

The results are discussed in Section 5 of the main report.

## Table B1 PSA results for period 1 (2040 – 2090)

| **Individual worker risk** | | | | |
|---|---|---|---|---|
| **Total risk ($y^{-1}$)** | **% of total risk** | | | **% by hazard** |
| | **UILW** | **SILW** | **HLW/SF** | |
| **$1\ 10^{-6}$** | 55 | 45 | < 1 | |
| **Ext** | 28 | << 1 | << 1 | 28 |
| **Imp** | 27 | 2 | 0 | 29 |
| **Fire** | << 1 | 40 | 0 | 40 |
| **Imp+fire** | < 1 | 3 | << 1 | 3 |
| **Individual public risk** | | | | |
| **Total risk ($y^{-1}$)** | **% of total risk** | | | **% by hazard** |
| | **UILW** | **SILW** | **HLW/SF** | |
| **$5\ 10^{-9}$** | 87 | 13 | <1 | |
| **Ext** | 4 | <<1 | <<1 | 4 |
| **Imp** | <1 | <<1 | 0 | <1 |
| **Fire** | <1 | 10 | 0 | 10 |
| **Imp+fire** | 82 | 3 | <1 | 86 |
| **Public dose-frequency ladder** | | | | |
| **Dose band (mSv)** | **Frequencies** | | | **% of BSO** |
| | **Total** | **BSO** | **BSL** | |
| < 0.1 | $5\ 10^{-2}$ | N/A | | N/A |
| 0.1 – 1 | $1\ 10^{-4}$ | $10^{-2}$ | 1 | 1 |
| 1 – 10 | $3\ 10^{-7}$ | $10^{-3}$ | $10^{-1}$ | <<1 |
| 10 -100 | 0 | $10^{-4}$ | $10^{-2}$ | 0 |
| 100 – 1000 | 0 | $10^{-5}$ | $10^{-3}$ | 0 |
| > 1000 | 0 | $10^{-6}$ | $10^{-4}$ | 0 |
| **Worker dose-frequency ladder** | | | | |
| **Dose band (mSv)** | **Frequencies** | | | **% of BSO** |
| | **Max** | **BSO** | **BSL** | |
| < 2 | $1\ 10^{-2}$ | N/A | | N/A |
| 2 – 20 | $7\ 10^{-3}$ | $10^{-3}$ | $10^{-1}$ | 700 |
| 20 – 200 | $7\ 10^{-4}$ | $10^{-4}$ | $10^{-2}$ | 700 |
| 200 - 2000 | 0 | $10^{-5}$ | $10^{-3}$ | 0 |
| > 2000 | 0 | $10^{-6}$ | $10^{-4}$ | 0 |

**Table B2 PSA results for period 2 (2080-2120)**

| | | | | |
|---|---|---|---|---|
| **Individual worker risk** | | | | |
| **Total risk ($y^{-1}$)** | **% of total risk** | | **% by hazard** | |
| | **DNLEU** | **HLW/SF** | | |
| **$8\ 10^{-9}$** | 73 | 27 | | |
| **Ext** | < 1 | 17 | 18 | |
| **Imp** | 71 | 0 | 71 | |
| **Fire** | <<1 | 0 | <<1 | |
| **Impact + fire** | 1 | 10 | 11 | |
| **Individual public risk** | | | | |
| **Total risk ($y^{-1}$)** | **% of total risk** | | **% by hazard** | |
| | **DNLEU** | **HLW/SF** | | |
| **$2\ 10^{-11}$** | 14 | 86 | | |
| **Ext** | < 1 | 6 | 6 | |
| **Imp** | 2 | 0 | 2 | |
| **Fire** | <<1 | 0 | <<1 | |
| **Impact + fire** | 12 | 80 | 92 | |
| **Public dose-frequency ladder** | | | | |
| **Dose band (mSv)** | **Frequencies** | | | **% of BSO** |
| | **Total** | **BSO** | **BSL** | |
| < 0.1 | $3\ 10^{-2}$ | N/A | | N/A |
| 0.1 – 1 | $1\ 10^{-7}$ | $10^{-2}$ | 1 | <<1 |
| 1 – 10 | $1\ 10^{-7}$ | $10^{-3}$ | $10^{-1}$ | <<1 |
| 10 -100 | 0 | $10^{-4}$ | $10^{-2}$ | 0 |
| 100 - 1000 | 0 | $10^{-5}$ | $10^{-3}$ | 0 |
| > 1000 | 0 | $10^{-6}$ | $10^{-4}$ | 0 |
| **Worker dose-frequency ladder** | | | | |
| **Dose band (mSv)** | **Frequencies** | | | **% of BSO** |
| | **Max** | **BSO** | **BSL** | |
| < 2 | $7\ 10^{-3}$ | N/A | | N/A |
| 2 – 20 | $7\ 10^{-8}$ | $10^{-3}$ | $10^{-1}$ | <<1 |
| 20 – 200 | $7\ 10^{-8}$ | $10^{-4}$ | $10^{-2}$ | <<1 |
| 200 - 2000 | 0 | $10^{-5}$ | $10^{-3}$ | 0 |
| > 2000 | 0 | $10^{-6}$ | $10^{-4}$ | 0 |

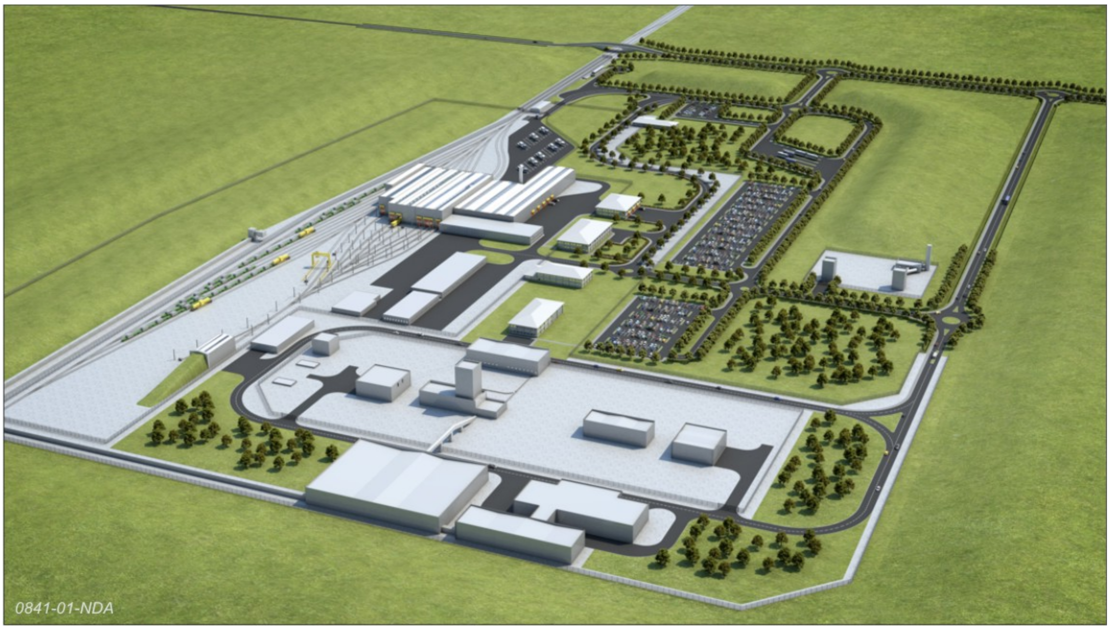
**Table B3 PSA results for period 3 (2120-2140)**

| **Individual worker risk** | | | |
|---|---|---|---|
| **Total risk ($y^{-1}$)** | **% of total risk** | | **% by hazard** |
| | **DNLEU** | **Pu/HEU** | |
| **$7\ 10^{-9}$** | 78 | 22 | |
| **Ext** | <1 | <<1 | <1 |
| **Imp** | 76 | 0 | 76 |
| **Fire** | <<1 | 0 | <<1 |
| **Impact + fire** | 2 | 22 | 24 |
| **Individual public risk** | | | |
| **Total risk ($y^{-1}$)** | **% of total risk** | | **% by hazard** |
| | **DNLEU** | **Pu/HEU** | |
| **$4\ 10^{-11}$** | 11 | 89 | |
| **Ext** | <<1 | <<1 | <1 |
| **Imp** | 11 | 0 | 11 |
| **Fire** | <<1 | 0 | <<1 |
| **Impact + fire** | <<1 | 89 | 89 |

| **Public dose-frequency ladder** | | | | |
|---|---|---|---|---|
| **Dose band (mSv)** | **Frequencies** | | | **% of BSO** |
| | **Total** | **BSO** | **BSL** | |
| < 0.1 | $3\ 10^{-2}$ | N/A | | N/A |
| 0.1 – 1 | 0 | $10^{-2}$ | 1 | 0 |
| 1 – 10 | $1\ 10^{-7}$ | $10^{-3}$ | $10^{-1}$ | <<1 |
| 10 -100 | 0 | $10^{-4}$ | $10^{-2}$ | 0 |
| 100 – 1000 | 0 | $10^{-5}$ | $10^{-3}$ | 0 |
| > 1000 | 0 | $10^{-6}$ | $10^{-4}$ | 0 |
| **Worker dose-frequency ladder** | | | | |
| **Dose band (mSv)** | **Frequencies** | | | **% of BSO** |
| | **Max** | **BSO** | **BSL** | |
| < 2 | $7\ 10^{-3}$ | N/A | | N/A |
| 2 – 20 | $7\ 10^{-8}$ | $10^{-3}$ | $10^{-1}$ | <<1 |
| 20 – 200 | $7\ 10^{-8}$ | $10^{-4}$ | $10^{-2}$ | <<1 |
| 200 - 2000 | $5\ 10^{-8}$ | $10^{-5}$ | $10^{-3}$ | <1 |
| > 2000 | 0 | $10^{-6}$ | $10^{-4}$ | 0 |

# Appendix C Engineering diagrams

## Figure C1 Example surface layout for a GDF




## Figure C2 Example underground layout for higher strength rock concept example

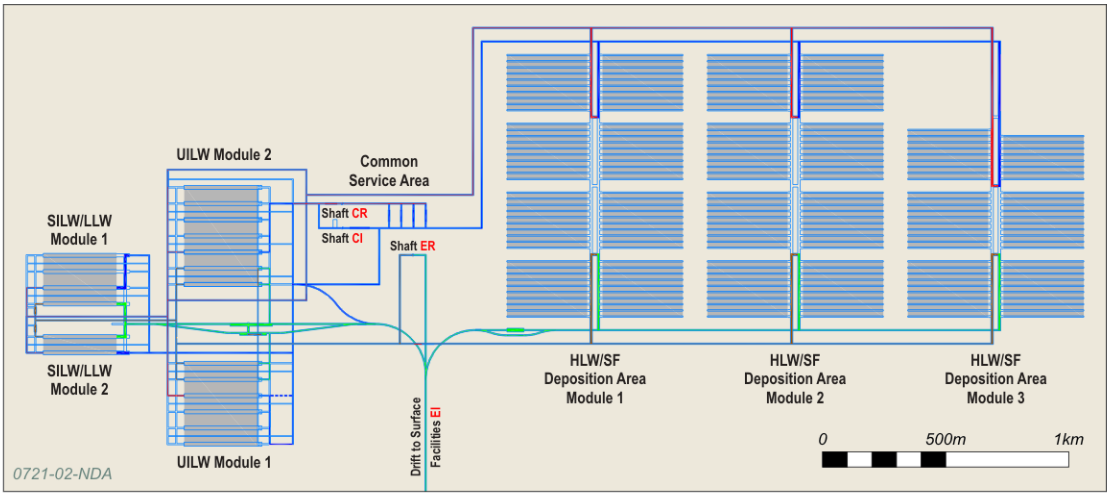
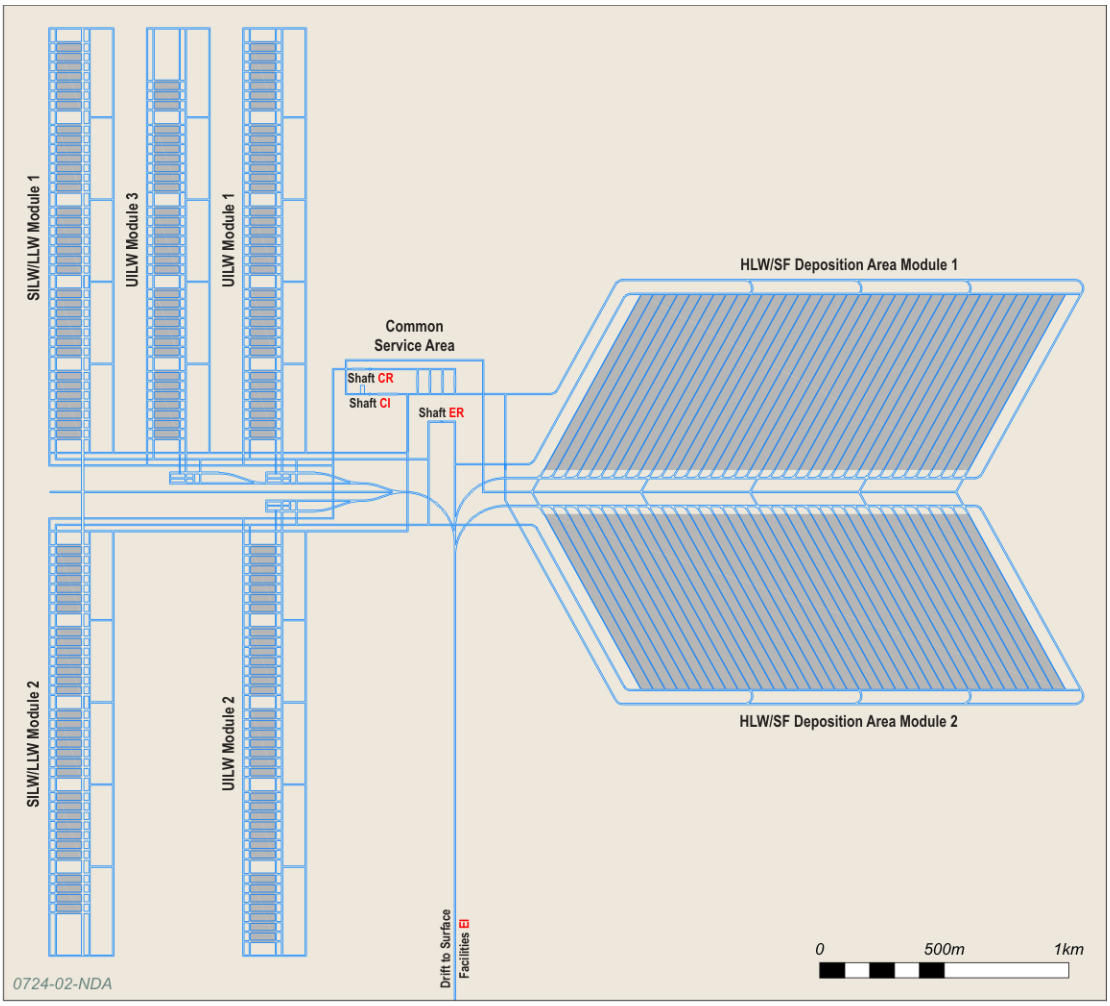
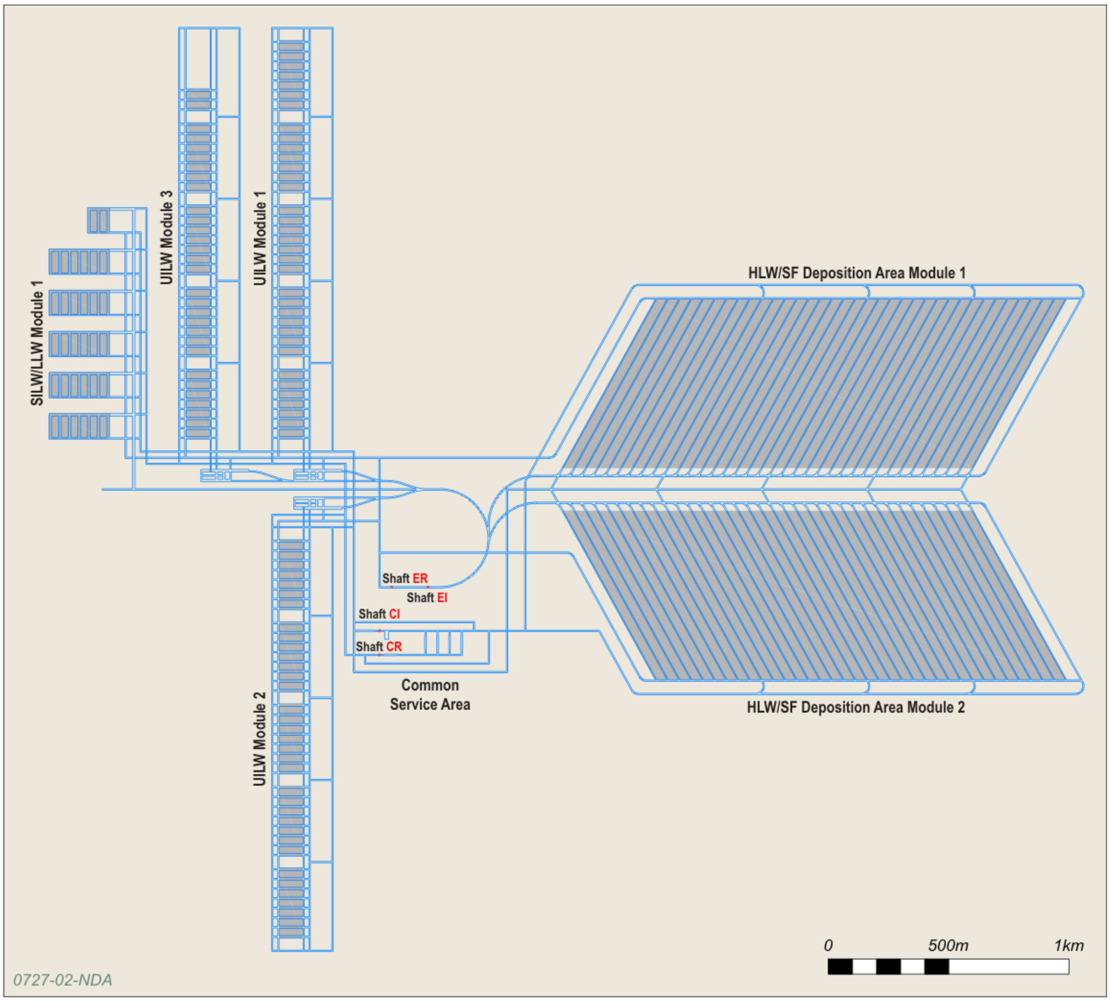


**Figure C3 Example underground layout for lower strength rock concept example**



## Figure C4 Example underground layout for evaporites concept example



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